U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.183, REVISION 1

Issue Date: October 2023
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ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable in complying with regulations for design basis accident (DBA) dose consequence analysis using an alternative source term (AST). This guidance for light-water reactor (LWR) designs includes the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an AST based in part on SAND2011-0128, "Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel," issued January 2011 (Ref. 1), and identifies significant attributes of other accident source terms that may be acceptable. The updated source terms reflect the current understanding of severe accidents and fission product behavior since the publication of RG 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000 (Ref. 2), and include low-burnup and high-burnup LWR fuels. The use of these source terms is not endorsed for mixed-oxide fuels. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the AST. Unusual site characteristics, plant design features, or other factors may require different assumptions, which the staff will consider on a case-by-case basis.

Applicability

This RG 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. 3), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 4), and to holders of renewed licenses under 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Ref. 5). In addition, although the guidance primarily reflects reviews of license amendment requests for light-water nuclear power plants licensed under 10 CFR Part 50, this RG could provide useful information to new reactor applicants and licensees under 10 CFR Part 50 or 10 CFR Part 52.

Written suggestions regarding this guide may be submitted through the NRC's public website in the NRC Library at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html, under Document Collections, in Regulatory Guides, at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html, and will be considered in future updates and enhancements to the "Regulatory Guide" series. During the development process of new guides, suggestions should be submitted within the comment period for immediate consideration. Suggestions received outside of the comment period will be considered if practical to do so or may be considered for future updates.

Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public website in the NRC Library at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html/ under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML23082A305. The regulatory analysis may be found in ADAMS under Accession No. ML21204A066. The associated draft guide DG-1389 may be found in ADAMS under Accession No. ML21204A065, and the staff responses to the public comments on DG-1389 may be found under ADAMS Accession No. ML23082A309

Applicable Regulations

- 10 CFR Part 50 provides regulations for licensing production and utilization facilities.

  - 10 CFR 50.67, "Accident source term," allows applicable licensees to voluntarily revise the accident source term used in design basis radiological consequence analyses. A licensee that seeks to revise its current accident source term shall apply for a license amendment under 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit." The regulation in 10 CFR 50.90 also allows the NRC to issue these amendments if the applicant's analysis demonstrates with reasonable assurance that certain dose reference values are met. In accordance with 10 CFR 50.67, a holder of an operating license issued before January 10, 1997, can voluntarily revise the accident source term used in design basis radiological consequence analyses. Applicants and licensees under 10 CFR Part 52 are not required to comply with 10 CFR 50.67; however, they may find that this guidance is useful in evaluations of fission product releases and the radiological consequences of LWR DBAs. Although the source term information is specific to LWR designs, this guide could provide useful information on radiological consequence analysis for non-LWR designs.

  - General Design Criterion (GDC) 19, "Control room," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCAs). It also states that adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions.

  - 10 CFR 50.34, "Contents of applications; technical information," requires, in part, that 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 (SSCs) of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. This regulation also requires applicants to provide an analysis of the proposed site.

- 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities.

  - 10 CFR 52.17, "Contents of applications; technical information," requires an early site permit applicant to provide a site safety analysis report that contains an analysis and evaluation of the major SSCs of the facility that bear significantly on the acceptability of the site and to evaluate the offsite radiological consequences from accidents.

  - 10 CFR 52.47, "Contents of applications; technical information," 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," 10 CFR 52.137, "Contents of applications; technical information," and 10 CFR 52.157, "Contents of applications; technical information in final safety analysis report," require applicants for standard design certification, combined license, standard design approval, and manufacturing license, respectively, to provide a final safety analysis report (FSAR) that, in part, presents a safety analysis of SSCs and to evaluate the offsite radiological consequences from accidents.

Related Guidance

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Ref. 6), provides guidance to the NRC staff for the review of safety analysis reports submitted as part of license applications for nuclear power plants.
    - SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," provides guidance to the NRC staff for reviewing radiological consequence analyses for LWRs using ASTs.
    - SRP Section 15.0.3, "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors," provides guidance to the NRC staff for reviewing radiological consequence analyses for new LWR applications, including advanced evolutionary and passive LWRs.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB), under control numbers 3150-0011 and 3150-0151, respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by email to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC, 20503; e-mail: oira_submissions@omb.eop.gov.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

TABLE OF CONTENTS

B. DISCUSSION

Reason for Revision

This revision of RG 1.183 (Revision 1) addresses new issues identified since the guide was originally issued. These include (1) using the term maximum hypothetical accident (MHA)[1] LOCA to clarify the accident that the staff finds acceptable to use to meet the description in the applicable regulations in section A above, with a clear delineation between source term assumptions and plant response,[2] (2) adding transient release fractions from empirical data from in-pile, prompt power pulse test programs and analyses from several international publications of fuel rod performance under prompt power excursion conditions, (3) revising steady-state release fractions for accidents other than the LOCA based on a revision to the American National Standards Institute/American Nuclear Society Standard 5.4, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel" (Ref. 7), (4) adding information to acknowledge that the RG may provide useful information for satisfying the radiological dose analysis requirements in 10 CFR Part 50 and 10 CFR Part 52 for new LWR applicants, including advanced evolutionary and passive LWR design and siting, (5) providing additional guidance for modeling boiling-water reactor (BWR) main steam isolation valve (MSIV) leakage, (6) adding guidance for accident tolerant fuel (ATF), high-burnup fuel, and increased enrichment source term analyses, (7) revising transport and decontamination models for the fuel handling DBA, (8) adding guidance for crediting hold-up and retention of MSIV leakage within the main steamlines and condenser for BWRs, and (9) providing additional meteorological assumption guidance.

Background

An accident source term is intended to represent a major accident involving significant core damage not exceeded by that from any other credible accident. NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. Facility-analyzed DBAs are not intended to be actual event sequences; rather, they are intended to be surrogates to enable deterministic evaluation of the response of engineered safety features (ESFs). These accident analyses are intentionally conservative to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion.

Probabilistic risk assessments (PRAs) can offer useful insights into system performance and suggest changes in how the desired defense in depth is achieved. The NRC's policy statement on the use of PRA methods (Ref. 8) calls for the use of PRA technology in all regulatory matters in a manner that complements the agency's deterministic approach and supports the traditional defense-in-depth philosophy, which continues to be an effective way to account for uncertainties in equipment and human performance.

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1 The MHA (also referred to as the maximum credible accident) is that accident whose consequences, as measured by the radiation exposure of the surrounding public, would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible. The MHA LOCA, as used in this guide, refers to a loss of core cooling resulting in substantial meltdown of the core with subsequent release into containment of appreciable quantities of fission products. These evaluations assume containment integrity with offsite hazards evaluated based on design basis containment leakage.

2 The MHA should be modeled with the deterministic substantial fuel melt source term being injected or overlaid into the radiological consequence analysis notwithstanding the operation of safety-related equipment designed to preclude significant fuel failure. The purpose of this approach would be to test the adequacy of the containment and other safety-related systems. Safety-related systems may be credited as described in Regulatory Position 5.1.2, as this designation ensures reliability in performing their safety function.

In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 9), which provides estimates of the accident source term that are more physically based and that could be applied to the design of new and advanced LWRs. NUREG-1465 presents a representative accident source term for a BWR and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment. The NRC staff considered the applicability of the revised source terms to operating reactors and determined that the current analytical approach, based on the source term in Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," issued March 1962 (Ref. 10), would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to reanalyze accidents using the revised source terms. However, the NRC staff determined that some operating reactor licensees might request to use an AST in analyses to support cost-beneficial licensing actions.

The NRC staff therefore initiated several actions to provide a regulatory basis for operating reactor licensees to use an AST[3] in design basis analyses. These initiatives resulted in the development and issuance of the final rule for 10 CFR 50.67 (64 FR 72001; December 23, 1999) and the subsequent issuance of RG 1.183, Revision 0, in July 2000, as implementing guidance for the rule.

Several RGs and SRP chapters describe the NRC's traditional methods for calculating the radiological consequences of DBAs. The staff developed that guidance to be consistent with the TID-14844 source term and the wholebody and thyroid dose guidelines stated in 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance" (Ref. 11). Many of those analysis assumptions and methods in that guidance are inconsistent with the AST and with the total effective dose equivalent (TEDE) criteria in 10 CFR 50.34, 10 CFR Part 52, and 10 CFR 50.67. This guide provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST.

RG 1.89, Revision 2, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," issued April 2023 (Ref. 12), offers guidance for equipment environmental qualification (EQ) using an AST. Reactor licensees licensed under 10 CFR Part 50 or applicants for licenses under 10 CFR Part 50 or 10 CFR Part 52 should use the applicable guidance in RG 1.89, Revision 2.

Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations when the method is applicable. However, consistent with the guidance in Revision 0 and Revision 1 of RG 1.183, if there are new or unreviewed issues, created by proposed site-specific implementations of an AST, or that conflict with the licensing basis or guidance (e.g., increase in burnup above the 62,000 megawatt days per metric ton of uranium of 5 weight-percent uranium-235 enrichment assumed in the Revision 0 methods), the licensee needs to ensure that the proposed implementation of the guidance is technically justified. Use of combinations and permutations of regulatory positions and assumptions from Revision 0 and Revision 1 will need appropriate technical justification and may require additional NRC review prior to approval.

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3 The NUREG-1465 source terms have often been referred to as the "revised source terms." In recognition that additional source terms may be identified in the future, 10 CFR 50.67 addresses "alternative source terms." This RG endorses a source term derived from SAND2011-0128 and provides guidance on the acceptable attributes of other ASTs.

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered IAEA Safety Requirements and Safety Guides pursuant to the Commission's International Policy Statement (Ref. 13) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 14).

The NRC staff did not identify any IAEA Safety Requirements or Guides with enough detailed information relevant for use by 10 CFR Part 50 and 10 CFR Part 52 licensees and applicants as related to the topic of this RG.

C. STAFF REGULATORY GUIDANCE

This section and the appendices to this RG contain regulatory positions that establish a method acceptable to the NRC staff for complying with regulations for DBA dose consequence analysis using an AST.

This RG applies to MHA LOCA models for applicants and licensees using zirconium-alloy cladded uranium dioxide (UO_2) fuel rod designs with reactor core burnups up to a maximum rod-average of 68 gigawatt-days per metric ton uranium (GWd/MTU) (and fuel enrichments up to 8 weight-percent uranium-235), including chromium-coated cladding (thicknesses less than 50 microns (μm)) and chromia-doped (up to 0.16 weight-percent) fuel. It is not applicable for other fuel-clad combinations, including fuel with iron-chromium-aluminum (FeCrAl) alloy cladding.

This RG applies to non-LOCA models for applicants and licensees with reactor core burnups up to a maximum rod-average burnup of 68 GWd/MTU (and fuel enrichments up to 8 weight-percent uranium-235) for currently approved (as of the issuance of this RG) zirconium-alloy cladded UO_2 fuel rod designs at power levels below the burnup-dependent power envelopes depicted in figure 1 of this guide.

1. Implementation of Accident Source Term

1.1 Generic Considerations

As used in this guide, the AST is an accident source term derived principally from SAND2011-0128; it differs from the TID-14844 and NUREG-1465 source terms used in the original and revised design and licensing of operating reactor facilities. ASTs may also be used by applicants for new LWRs, including advanced evolutionary and passive LWRs, under 10 CFR Part 50 and 10 CFR Part 52, and for existing operating reactor licensees under 10 CFR 50.34 and 10 CFR 50.67. This guide identifies an AST that is acceptable to the NRC staff and describes significant characteristics of other source terms that may be found acceptable. While the staff recognizes several potential uses of an AST, it is not possible to foresee all possible uses. Licensees may pursue technically justifiable uses of the ASTs in the most flexible manner in license amendments so long as these uses are compatible with maintaining a clear, logical, and consistent design basis and continue to comply with NRC regulations. These license amendment requests should demonstrate that the facility, as modified, will continue to provide sufficient safety margins, with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs.

1.1.1 Safety Margins

Licensees should evaluate their proposed uses of this guide, and the associated proposed facility modifications and changes to procedures, to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. For example, changes, or the net effect of multiple changes, that result in a reduction in safety margins may require prior NRC approval. If the initial AST implementation, consistent with the guidance in RG 1.183, Revision 1, is approved by the staff and becomes part of the facility design basis, licensees may use 10 CFR 50.59, "Changes, tests and experiments," and its supporting guidance to assess facility modifications and changes to procedures that are described in the updated FSAR.

1.1.2 Defense in Depth

Licensees should evaluate their proposed uses of an AST, and the associated proposed facility modifications and changes to procedures, to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected accident frequency, consequences of challenges to the system, and uncertainties. For facilities to which the GDC (see Appendix A to 10 CFR Part 50) apply, compliance with these criteria is required. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities (e.g., reliance on manual operator actions, use of potassium iodide as a prophylactic drug) or self-contained breathing apparatus or post-accident entries into vital areas to maintain required equipment qualifications.

Licensees should evaluate proposed modifications that seek to downgrade or remove required engineered safeguards equipment, to confirm that the modifications do not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program.

1.1.3 Integrity of Facility Design Basis

The DBA source term used for dose consequence analyses is a fundamental assumption and the basis for much of the facility design. Additionally, many aspects of an operating reactor facility are derived from the radiological design analyses that incorporated the TID-14844 accident source term. Although a complete reassessment of all facility radiological analyses would be desirable, the NRC staff determined that recalculation of all design analyses for operating reactors would generally not be necessary. Regulatory Position 1.3 of this guide contains guidance on which analyses should be updated as part of the AST implementation submittal and which may need to be updated in the future as additional modifications are made.

This approach for operating reactors creates two tiers of analyses—one based on the previous TID-14844 source term, and one based on an AST. The radiological acceptance criteria would also differ, as some analyses are based on whole-body and thyroid criteria and some are based on TEDE criteria. Full implementation of an AST revises the plant licensing basis to specify the AST in place of the previous TID-14844 accident source term and establishes the TEDE dose as the new acceptance criteria. Selective implementation of an AST also revises the plant licensing basis and may establish the TEDE dose as the new acceptance criteria. Selective implementation differs from full implementation only in the scope of the change. In either case, the facility design bases should clearly indicate that the source term assumptions and radiological criteria in the affected analyses have been superseded and that future revisions of these analyses, if any, will use the updated approved assumptions and criteria.

Radiological analyses generally should be based on assumptions and inputs that are consistent with corresponding data used in other design basis safety analyses unless the use of these data would result in nonconservative results or otherwise conflict with regulatory guidance.

1.1.4 Emergency Preparedness Applications

The regulations in 10 CFR 50.47, "Emergency plans," include the requirements for emergency preparedness at nuclear power plants. Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," to 10 CFR Part 50 states additional requirements. NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," issued December 1978 (Ref. 15), includes the planning basis for many of these requirements.[4] This joint effort by the U.S. Environmental Protection Agency (EPA) and the NRC considered the principal characteristics (such as nuclides released and distances) likely to be involved in a spectrum of design basis and severe (core melt) accidents. No single accident scenario is the basis of the required emergency preparedness. The objective of the planning is to provide public protection encompassing a wide spectrum of possible events, with a sufficient basis for extension of response efforts for unanticipated events. The NRC and EPA issued these requirements after a long period of involvement by many stakeholders, including the Federal Emergency Management Agency, other Federal agencies, local and State governments (and in some cases foreign governments), private citizens, utilities, and industry groups.

Although the NRC based the AST in this guide on a limited spectrum of severe accidents, the particular characteristics are tailored specifically for use in DBA analysis. The AST is not representative of the wide spectrum of possible events that make up the planning basis of emergency preparedness. Therefore, the AST is insufficient by itself as a basis for requesting relief from the emergency preparedness requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50.

This guidance does not, however, preclude the appropriate use of insights based on the AST in establishing emergency response procedures, such as those associated with emergency dose projections, protective measures, and severe accident management guides.

1.1.5 Applicability to New Light-Water Reactor Applications, Including Advanced Evolutionary and Passive Designs

The NRC originally created RG 1.183 for use by existing nuclear power reactors to satisfy regulations under 10 CFR 50.34 and 10 CFR 50.67. This revision of RG 1.183 extends the applicability of the proposed RG for use by new LWR power reactor applications, including advanced evolutionary and passive LWR designs, in satisfying the radiological dose analysis requirements in 10 CFR Part 50 and 10 CFR Part 52 for safety and siting analyses. New reactor applicants and licensees may use the guidance in this RG that is applicable to their design to meet the accident radiological consequence analysis requirements in 10 CFR Part 50 or 10 CFR Part 52 for permits, licenses, approvals, or certifications. To review these applications, the staff will use the methodology and assumptions stated in this RG that apply to the design.

Both Revision 0 and Revision 1 of RG 1.183 will be available for use, because Revision 0 will not be withdrawn. Each revision provides a method acceptable to the NRC for compliance with the applicable regulations specified in the guidance. The use of combinations and permutations of regulatory positions and assumptions from Revision 0 and Revision 1 will need appropriate technical justification and may require additional NRC review prior to approval. For example, if a licensee wants to move to a rod-average burnup limit of 68 GWd/MTU and enrichments up to 8 weight-percent uranium-235 for certain near-term ATF designs, guidance in RG 1.183, Revision 1, and applicable appendices would be used since RG 1.183, Revision 0, is approved for up to 62 GWd/MTU.

1.2 Scope of Implementation

The AST described in this guide is characterized by radionuclide composition and magnitude, the chemical and physical form of the radionuclides, and the timing of their release. The accident source term is a fundamental assumption and the basis of much of the facility design.

4 NUREG-0654, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," issued November 1980 (Ref. 16), also addresses this planning basis.

For operating reactors to which 10 CFR 50.67 applies, a complete implementation of an AST would upgrade all existing radiological analyses and would consider the impact of all five characteristics of a source term as defined in 10 CFR 50.2, "Definitions." However, the NRC staff has determined that there could be implementations for which this level of reanalysis may not be necessary. For holders of operating licenses, as defined in the applicability section of 10 CFR 50.67, two categories of AST implementation are defined: full and selective. Regulatory Positions 1.2.1 and 1.2.2 describe these categories.

For the radiological consequence analyses for new reactor permit, license, approval, or certification applications (e.g., those under 10 CFR 50.34(a)(1) or 10 CFR Part 52), the accident source term should consider all characteristics of a source term as defined in 10 CFR 50.2 and detailed in Regulatory Position 2 of this guide. Full and selective implementations, as used in the regulatory positions that follow, do not apply to new reactor applicants.

1.2.1 Full Implementation

Full implementation is a modification of the facility design basis that addresses all characteristics of the AST: specifically, the composition and magnitude of the radioactive material, its chemical and physical form, and the timing of its release. Full implementation revises the plant licensing basis to specify the AST in place of the previous accident source term and establishes the TEDE as the new acceptance criterion. This applies not only to the analyses performed in the application (which may include only a subset of the plant analyses), but also to all future design basis analyses. A full implementation is considered when, at a minimum, the application includes a reanalyzed MHA LOCA using the guidance in appendix A to this guide. Regulatory Position 1.3 in this guide provides additional guidance on the analysis. Since the AST and the TEDE criteria will become part of the facility design basis, new applications of the AST will not require prior NRC approval unless the new application involves a change to a technical specification, or unless the licensee's 10 CFR 50.59 evaluation concludes that prior NRC approval is required. However, a change from an approved AST to a different AST that is not approved for use at that facility would require a license amendment under 10 CFR 50.67.

1.2.2 Selective Implementation

Selective implementation is a modification of the facility design basis that (1) is based on one or more of the characteristics of the AST or (2) entails reevaluation of a limited subset of the design basis radiological analyses. The NRC staff will allow licensees flexibility in adopting technically justified selective implementations, provided that they maintain a clear, logical, and consistent design basis. An example of an application of selective implementation would be one in which a licensee desires to use the release timing insights of the AST to increase the required closure time for a containment isolation valve by a small amount. Another example would be a request to remove the charcoal filter media from the spent fuel building ventilation exhaust. In the latter example, the licensee may need to reanalyze only DBAs that credited the iodine removal performed by the charcoal media. Regulatory Position 1.3 of this guide provides additional analysis guidance. NRC approval for the AST (and the TEDE dose criterion) will be limited to the particular selective implementation proposed by the licensee. The licensee would be able to make subsequent modifications to the facility and changes to procedures based on the selected AST characteristics incorporated into the design basis under the provisions of 10 CFR 50.59. However, the use of other AST characteristics or the use of TEDE criteria that are not part of the approved design basis, and changes to previously approved AST characteristics would require prior staff approval under 10 CFR 50.67. As an example, a licensee with an implementation involving only timing, such as relaxed closure time on isolation valves, could not use 10 CFR 50.59 as a mechanism to implement a modification involving a reanalysis of the MHA LOCA. However, the licensee could extend use of the timing characteristic to adjust the closure time on isolation valves not included in the original approval.

1.3 Scope of Required Analyses

1.3.1 Design Basis Radiological Analyses

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:

a. EQ of equipment (10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants");

b. control room habitability (GDC 19 of Appendix A to 10 CFR Part 50);

c. emergency response facility habitability (paragraph IV.E.8 of Appendix E to 10 CFR Part 50);

d. AST (10 CFR 50.67);

e. environmental reports (10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions") (Ref. 17);

f. 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);[5] and

g. power reactor applications for early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses (10 CFR Part 52), construction permits (10 CFR 50.34(a)), or operating licenses (10 CFR 50.34(b)).

There may be other areas in which the technical specification bases and various licensee commitments refer to evaluations that use an AST. A plant's licensing bases may also include, but are not limited to, the following sections of NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980 (Ref. 18):

a. post-accident access shielding (II.B.2),
b. post-accident sampling capability (II.B.3),
c. accident monitoring instrumentation (II.F.1),
d. leakage control (III.D.1.1),
e. emergency response facilities (III.A.1.2), and
f. control room habitability (III.D.3.4).

For applications under 10 CFR Part 52 (e.g., 10 CFR 52.47(a)(8)), 10 CFR 50.34(f) requires that each applicant for a design certification, design approval, combined license, or manufacturing license shall demonstrate compliance with the technically relevant portions of the requirements related to the accident at the Three Mile Island nuclear reactor in 10 CFR 50.34(f)(1)–(3), except for 10 CFR 50.34(f)(1)(xii), 10 CFR 50.34(f)(2)(ix), and 10 CFR 50.34(f)(3)(v). These requirements include the NUREG-0737 sections listed above.

5 For licensees that have implemented an AST, the dose guidelines of 10 CFR 50.67 supersede those of 10 CFR 100.11.

1.3.2 *Reanalysis Guidance*

Any full or selective implementation of an AST and any associated facility modification should be supported by an evaluation of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above, as well as any other facility-specific requirements. These impacts may be caused by (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the reevaluation will necessarily depend on the specific facility modification proposed[6] and on whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and design bases appropriately. The NRC considers an analysis to be affected if the proposed modification changes one or more assumptions or inputs used in the analysis, in such a way that the results, or the conclusions drawn from the results, are no longer valid. A licensee may use NRC-approved generic analyses, such as those performed by owner groups or vendor topical reports, provided that the licensee justifies the applicability of the generic conclusions to the specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be recalculated, the licensee should update all affected assumptions and inputs and address all selected characteristics of the AST and the TEDE criteria. Any license amendment request should describe the licensee's reanalysis effort and provide statements on the acceptability of the proposed implementation, including modifications, with respect to each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.

The NRC staff has evaluated the impact of the AST on three representative operating reactors (Ref. 19). This evaluation determined that radiological analysis results based on the TID-14844 source term assumptions and the whole-body and thyroid methodology generally bound the results from analyses based on the AST and TEDE methodology. Licensees may use the applicable conclusions of this evaluation in addressing the impact of the AST on design basis radiological analyses. However, this does not exempt licensees from evaluating the remaining radiological and non-radiological impacts of the AST implementation and the impacts of the associated plant modifications. For example, a selective implementation based on the timing insights of the AST may change the required isolation time for the containment purge dampers from 2.5 seconds to 5.0 seconds. This application may be acceptable without dose calculations. However, the licensee may need to evaluate the ability of the damper to close against increased containment pressure or the ability of ductwork downstream of the dampers to withstand increased stresses.

An application for full implementation is considered when, at a minimum, the application includes a reanalyzed MHA LOCA using the guidance in appendix A to this guide. The licensee should update other design basis analyses in accordance with the guidance in this section.

A selective implementation of an AST and of any associated facility modification based on the AST should evaluate all the radiological and nonradiological impacts of the proposed actions as they apply to the particular implementation. The licensee should update design basis analyses in accordance with the guidance in this section. There is no requirement that an MHA LOCA analysis be performed. The analyses performed need to address all impacts of the proposed modification, the selected characteristics of the AST, and if dose calculations are performed, the TEDE criteria. For selective

6 For example, a proposed modification to change the timing of a containment isolation valve from 2.5 seconds to 5.0 seconds might be acceptable without any dose calculations. However, a proposed modification that would delay containment spray actuation could involve recalculation of MHA LOCA doses, reassessment of the containment pressure and temperature transient, recalculation of sump pH, reassessment of the emergency diesel generator loading sequence, integrated doses to equipment in the containment, and more.

implementations based on the timing characteristic of the AST (e.g., a change in the closure timing of a containment isolation valve), reanalysis of radiological calculations may not be necessary if the modified elapsed time remains a fraction (e.g., 0.25) of the time between accident initiation and the onset of the gap release phase. Longer time delays may be considered on an individual basis. For longer time delays, it may be necessary to evaluate the radiological consequences and other impacts of the delay, such as blockage by debris in sump water. If affected design basis analyses are to be recalculated, all affected assumptions and inputs should be updated, and all selected characteristics of the AST and the TEDE criteria should be addressed.

1.3.3 Use of Sensitivity or Scoping Analyses

It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a sensitivity analysis is an evaluation that considers how the overall results vary as an input parameter (in this case, an AST characteristic) is varied, for a given set of assumptions. A scoping analysis is a brief evaluation that uses conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but that are otherwise largely based on generic assumptions and inputs. Such cases might include postaccident vital area access dose calculations, shielding calculations, and equipment EQ (integrated dose). It may be possible to identify a bounding case, reanalyze that case, and use the results to draw conclusions about the remaining analyses. It may also be possible to show that, for some analyses, the whole-body and thyroid doses determined with the previous source term would bound the TEDE obtained using the AST. Where present, arbitrary "designer margins" may be adequate to bound any impact of the AST and the TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should discuss the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low-population zone (LPZ), or control room dose unless a clear and defensible basis exists for their doing so. Sensitivity analyses should avoid including well-defined parameters, such as atmospheric dispersion factors, that are based on site-specific data. Sensitivity studies used for the purpose of identifying a bounding design basis case should not vary parameters that are not a part of the licensing basis.

1.3.4 Updating Analyses Following Implementation

Full implementation of the AST replaces the previous accident source term with the approved AST and the TEDE criteria for all design basis radiological analyses. The implementation may have been supported in part by sensitivity or scoping analyses that concluded that many of the design basis radiological analyses would remain bounding for the AST and the TEDE criteria and would not require updating. After the implementation is complete, there may be a subsequent need (e.g., a planned facility modification) to revise these analyses or to perform new analyses. For these recalculations, the NRC staff expects that all affected analyses will address all characteristics of the AST and the TEDE criteria incorporated into the design basis. Reevaluation using the previously approved source term may not be appropriate. Since the AST and the TEDE criteria are part of the approved design basis for the facility, use of the AST and TEDE criteria in new applications at the facility does not constitute a change in analysis methodology that would require NRC approval.[7]

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7 In performing screenings and evaluations pursuant to 10 CFR 50.59, it may be necessary to compare dose results (figure of merit) expressed in terms of whole-body and thyroid with results expressed in terms of TEDE. Either figure of merit represents different systems of dosimetry. There is no methodology that converts the figures of merit between systems. Therefore, to calculate the desired figure of merit, the appropriate dosimetry methodology (i.e., dose conversion factors) must be applied.

This guidance also applies to selective implementations, to the extent that the affected analyses are within the scope of the approved implementation as described in the facility design basis. In these cases, the updated analyses should consider the characteristics of the AST and the TEDE criteria identified in the facility design basis. Use of other characteristics of the AST or TEDE criteria that are not part of the approved design basis and changes to previously approved AST characteristics require prior NRC staff approval under 10 CFR 50.67.

1.3.5 Equipment Environmental Qualification

A proposed plant modification associated with the AST implementation may affect current EQ analyses. To address these impacts, the licensee should update EQ analyses that have assumptions or inputs affected by the plant modification. New facilities proposing to implement an AST should use the guidance in RG 1.89, Revision 2.

1.4 Risk Implications

This guide provides regulatory assumptions that licensees should use in calculating the radiological consequences of DBAs. These assumptions have no direct influence on the probability of the design basis initiator. These analysis assumptions cannot increase the core damage frequency or the large early release frequency. However, facility modifications made possible by the AST could have an impact on risk. If the proposed implementation of the AST involves changes to the facility design that would invalidate assumptions made in the facility's PRA, the licensee should evaluate the impact on the existing PRAs.

Evaluations should consider the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is reason to question adequate protection of public health and safety.

The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. Guidance appears in RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued January 2018 (Ref. 20). Additional information on how the staff performs integrated risk-informed decision-making for licensing reviews is available in LIC-206, "Integrated Risk-Informed Decision-Making for Licensing Reviews," dated June 26, 2020 (Ref. 21).

1.5 Submittal Requirements and Information

According to 10 CFR 50.90, an application for an amendment must fully describe the changes desired and should follow, as far as applicable, the form prescribed for original applications. RG 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," issued November 1978 (Ref. 22), provides additional guidance. As stated in 10 CFR 50.67, "[t]he NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that" the identified dose criteria are met. It is the licensee's analyses that will become part of the design and licensing basis of the facility are key aspects of the submittal information. This information submitted in the amendment request, along with the current plant design as documented in the FSAR, staff safety evaluation reports, regulatory guidance, other licensee commitments, and staff experience gained in approving similar requests for other plants also provide important information. The methods and analyses described in this guidance are acceptable to the NRC staff for meeting the requirements in 10 CFR 50.67. However, if licensees propose to use alternatives different from this guidance, licensees should provide sufficient justification to demonstrate that the requirements for the pertinent regulations are satisfied. Importantly, as noted above and stated in 10 CFR 50.67, the amendment may be issued only "if the applicant's analysis demonstrates with reasonable assurance" that the identified dose criteria are met, and therefore, licensees should ensure that they present adequate information, such as analysis assumptions, inputs, and methods, in the submittal to support the staff's assessment.[8]

The amendment request should describe the licensee's analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. Consistent with 10 CFR 50.90, the licensee shall, as far as applicable, follow the form prescribed for original applications. Typically, original applications include FSAR pages and technical specifications. Licensees should submit affected FSAR pages and technical specifications annotated with changes that reflect the revised analyses. Additionally, the NRC staff recommends that licensees submit the actual calculation documentation. In lieu of submitting marked-up FSAR pages, licensees should include a detailed list, preferably in tabular format, of all changes being proposed between the current facility licensing basis and the requested license amendment, as well as appropriate justification for each change.

If the licensee has used a currently approved version of an NRC-sponsored computer code, the NRC staff's review and confirmatory analyses will often be more efficient if the licensee identifies the code used and submits the inputs used in the calculations performed using that code. In many cases, this information will help support efficiency in the NRC staff's review and confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.

Applications for licenses, certifications, and approvals under 10 CFR Part 52 have requirements similar to those stated above for license amendment submittals. RG 1.206, Revision 1, "Applications for Nuclear Power Plants," issued October 2018 (Ref. 23), provides additional guidance on combined license applications.

1.6 Final Safety Analysis Report Requirements

The regulations in 10 CFR 50.71, "Maintenance of records, making of reports," include the requirements for updating the facility's FSAR. Specifically, 10 CFR 50.71(e) requires that the FSAR be updated to include all changes made in the facility or in procedures described in the FSAR, as well as all safety analyses and evaluations performed by the licensee in support of approved requests for license amendments or in support of conclusions that changes did not require a license amendment in accordance with 10 CFR 50.59. The analyses required by 10 CFR 50.67 are subject to this requirement. The licensee should update the affected radiological analysis descriptions in the FSAR to reflect the design basis changes to the methodology and input. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numerical results. RG 1.70 provides additional guidance. The licensee should remove the descriptions of superseded analyses from the FSAR in the interest of maintaining a clear design basis.

2. Attributes of an Acceptable Accident Source Term

The NRC did not provide an acceptable AST in 10 CFR 50.67. Regulatory Position 3 of this guide identifies an AST that is acceptable to the NRC staff for use in new power reactor applications and operating power reactors. The NRC, its contractors, various national laboratories, peer reviewers, and others have expended substantial effort in performing severe accident research and in developing the source terms in SAND2011-0128. However, future research may identify opportunities for changes in

8 The analyses required by 10 CFR 50.67 are important to the design basis of the facility, and 10 CFR 50.34 requires design basis safety analyses and evaluations; they are a significant input to the evaluations required by 10 CFR 50.92, "Issuance of amendment," or 10 CFR 50.59.

these source terms. The NRC staff will consider applications for an AST different from that identified in this guide, although the staff does not expect to approve any MHA LOCA source term that is not of the same quality as the source terms in NUREG-1465 and SAND2011-0128. An acceptable AST has the following attributes:

a. The AST is based on major accidents hypothesized for the purposes of design analyses, or on consideration of possible accidental events that could result in hazards not exceeded by those from other accidents considered credible. The AST addresses events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products.

b. The AST is expressed in terms of the times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

c. The AST is not based on a single accident scenario but instead represents a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. However, risk insights alone are not an acceptable basis for excluding a particular event. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

d. The AST has a defensible technical basis supported by sufficient experimental and empirical data, is verified and validated, and is documented in a scrutable form that facilitates public review and discourse.

e. The AST is peer reviewed by appropriately qualified subject-matter experts. The peer review comments, and their resolution should be part of the documentation supporting the AST.

Regulatory Position 3 of this guide also identifies steady-state fission product release fractions residing in the fuel rod void volume (plenum and pellet-to-cladding gap) that are acceptable to the NRC staff for use in new power reactor applications and operating power reactors. As an alternative to these bounding release fractions, appendix I to this guide provides an acceptable analytical procedure for calculating plant-specific or fuel-rod-design-specific fission product release fractions. The NRC internal memorandum “Technical Basis for Draft RG 1.183 Revision 1 (2021) Non-LOCA Fission Product Release Fractions,” dated July 28, 2021 (Ref. 24), provides an example calculation illustrating the application of this analytical procedure.

3. Accident Source Term[9]

This regulatory position provides an AST that is acceptable to the NRC staff. It offers guidance on the fission product inventory, release fractions, timing of the release phases, radionuclide composition, chemical form, and fuel damage for LOCA and non-LOCA DBAs. The data in Regulatory Positions 3.1 through 3.5 are fundamental to the definition of an AST. Once approved, the AST assumptions or parameters specified in these positions become part of the facility’s design basis. The NRC will evaluate deviations from this guidance against Regulatory Position 2. After the NRC staff has approved an implementation of an AST, subsequent changes to the AST will also require NRC staff review under 10 CFR 50.67.

9 The data in this Regulatory Position do not apply to cores containing mixed oxide fuel or to near-term ATF FeCrAl and long-term ATF concepts.

3.1 Fission Product Inventory

The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full-power operation of the core with, as a minimum, currently licensed values for fuel enrichment and fuel burnup, and an assumed core power equal to the currently licensed rated thermal power times the approved core power measurement uncertainty factor (e.g., 1.02). These parameters should be examined to maximize fission product inventory. For non-LOCA DBAs, the NRC staff will consider on a case-by-case basis the use of more explicit methods to calculate the fission product inventory and reactor coolant system activity, such as using the burnup, power history, and peaking factor from the accident analyses (e.g., chapter 14 or 15) in the updated FSAR, for each rod predicted to fail. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach maximum values.[10] The core inventory should be determined using an appropriate computer code for calculating isotope generation and depletion. The core inventory factors (in curies per megawatt thermal) provided in TID-14844 and used in some analysis computer codes were derived for low-burnup, low-enrichment fuel and should not be used with higher burnup or higher enrichment fuels. The code should model the fuel geometries, material composition, and burnup, and the cross section libraries used should be applicable to the projected fuel burnup.

For the MHA LOCA, all fuel assemblies in the core are assumed to be affected, and the analysis should use the core-average inventory. For DBA events that do not involve the entire core, the fission product inventory of each damaged fuel rod is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, the analysis should apply the radial peaking factors (for PWRs, these are contained in the facility’s core operating limits report or technical specifications) in determining the inventory of the damaged rods.

The licensee should not adjust the fission product inventory for events postulated to occur during power operations at less than full-rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shut down (e.g., a fuel handling accident), the licensee may model radioactive decay from the time of shutdown.

3.2 Release Fractions

For the MHA LOCA, table 1 (for BWRs) and table 2 (for PWRs) in this RG list the core inventory release fractions, by radionuclide group, for the gap release and early in-vessel damage phases. These fractions are applied to the maximum core inventory described in Regulatory Position 3.1.

The limitations on the use of tables 1 and 2 in this RG are based on several reference documents. First, tables 1 and 2 are based on accident source terms from SAND2011-0128 that use the maximum release fractions from low- and high-burnup results. The tables 1 and 2 source terms were derived by examining a set of accident sequences for current LWR designs; they reflect the current understanding of severe accidents and fission product behavior since the publication of NUREG-1465. The use of the maximum release fractions from low- and high-burnup results accounts for different radionuclide quantities at different burnups throughout the operating cycle. Second, the NRC internal memorandum “Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment,” dated May 13, 2020 (Ref. 25), in part, and insights from a literature review on ATFs reported in NUREG/CR-7282, “Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases,” issued July 2021 (Ref. 26), support the applicability of SAND2011-0128 for licensees using zirconium-alloy cladded uranium dioxide (UO2) fuel rod designs

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10 Note that for some radionuclides, such as cesium-137, maximum values will not be reached before fuel offload. Thus, the maximum inventory at the end of life should be used.

with reactor core burnups up to a maximum rod-average of 68 GWd/MTU (and fuel enrichments up to 8 weight-percent uranium-235), including chromium-coated cladding (thicknesses less than 50 µm) and chromia-doped (up to 0.16 weight-percent) fuel. SAND2011-0128 is not applicable for other fuel-clad combinations, including fuel with FeCrAl alloy cladding. Lastly, the NRC internal memorandum "Letter Report on Evaluation of the Impact of Fuel Fragmentation, Relocation, and Dispersal for the Radiological Design Basis Accidents in Regulatory Guide 1.183," dated July 20, 2021 (Ref. 27), assesses the impact of fuel fragmentation, relocation, and dispersal behavior on the accident source terms from SAND2011-0128. Based on information provided in that memo, for the purposes of assessing the radiological consequences of the MHA LOCA, the impact of fuel fragmentation, relocation, and dispersal does not need to be considered for the range of applicability of burnups and enrichments in SAND2011-0128. This RG does not provide guidance related to an acceptable treatment of fuel dispersal during non-LOCA DBAs.

Table 1. BWR Core Inventory Fraction Released into Containment Atmosphere

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	0.008	0.96	0.968
Halogens	0.003	0.54	0.543
Alkali Metals	0.003	0.14	0.143
Tellurium Metals	0.003	0.39	0.393
Barium, Strontium	0.00	0.005	0.005
Noble Metals	0.00	2.7×10^{-3}	2.7×10^{-3}
Cerium Group	0.00	1.6×10^{-7}	1.6×10^{-7}
Lanthanides	0.00	2.0×10^{-7}	2.0×10^{-7}
Molybdenum	0.00	0.03	0.03

Table 2. PWR Core Inventory Fraction Released into Containment Atmosphere

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	0.022	0.94	0.962
Halogens	0.007	0.37	0.377
Alkali Metals	0.005	0.23	0.235
Tellurium Metals	0.007	0.30	0.307
Barium, Strontium	1.4×10^{-3}	0.004	5.4×10^{-3}
Noble Metals	0.00	0.006	0.006
Cerium Group	0.00	1.5×10^{-7}	1.5×10^{-7}
Lanthanides	0.00	1.5×10^{-7}	1.5×10^{-7}
Molybdenum	0.00	0.10	0.10

For non-LOCA DBAs, table 3 (for BWRs) and table 4 (for PWRs) list the maximum steady-state fission product release fractions residing in the fuel rod void volume (plenum and pellet-to-cladding gap), by radionuclide group, available for release upon cladding breach. The licensing bases of some facilities may include non-LOCA events that assume the release of the gap activity from the entire core. For events involving the entire core, the core-average gap fractions of tables 1 and 2 may be used, and the radial peaking factor may be omitted.

The applicability of the steady-state fission product release fractions in tables 3 and 4 is limited to currently approved (as of the issuance of this RG) full-length UO_2 fuel rod designs operating up to a maximum rod-average burnup of 68 GWd/MTU at power levels below the burnup-dependent power envelopes depicted in figure 1. In figure 1, the bounding rod-average power refers to the rod-average linear heat generation rate of the peak rod, and the peak power refers to the maximum local linear heat generation rate in the core. Licensees should make adjustments to account for power uncertainties and plant maneuvering when comparing operating power histories to figure 1. If it can be demonstrated that local power level, rate of fission gas release, and cumulative fission gas release remain less than those of the limiting co-resident UO_2 fuel rod, then the steady-state fission product release fractions in tables 3 and 4 apply to fuel rod designs containing integral burnable absorbers (e.g., gadolinia). One acceptable means of demonstrating this is by using an NRC-approved fuel performance code that has fission gas release models that are applicable to the integral burnable absorber fuel designs. If BWR part-length fuel rods are treated as full-length fuel rods with respect to overall quantity of fission products and are operated below the burnup-dependent BWR peak power envelope in figure 1, then table 3 steady-state fission product release fractions apply to these part-length fuel rod designs. Applicability to future fuel rod designs, including Cr-coated zirconium (Zr) cladding, non-Zr claddings, doped UO_2 fuel, high-density fuel, and mixed-oxide fuel, will be evaluated on a case-by-case basis. Appendix I provides an acceptable analytical technique for calculating plant-specific or fuel-rod-design-specific fission product release fractions.

Table 3. BWR Steady-State Fission Product Release Fractions Residing in the Fuel Rod Plenum and Gap

Group	Fraction
I-131	0.03
I-132	0.03
Kr-85	0.32
Other Noble Gases	0.03
Other Halogens	0.02
Alkali Metals	0.16

Table 4. PWR Steady-State Fission Product Release Fractions Residing in the Fuel Rod Plenum and Gap

Group	Fraction
I-131	0.07
I-132	0.07
Kr-85	0.40
Other Noble Gases	0.06
Other Halogens	0.04
Alkali Metals	0.20

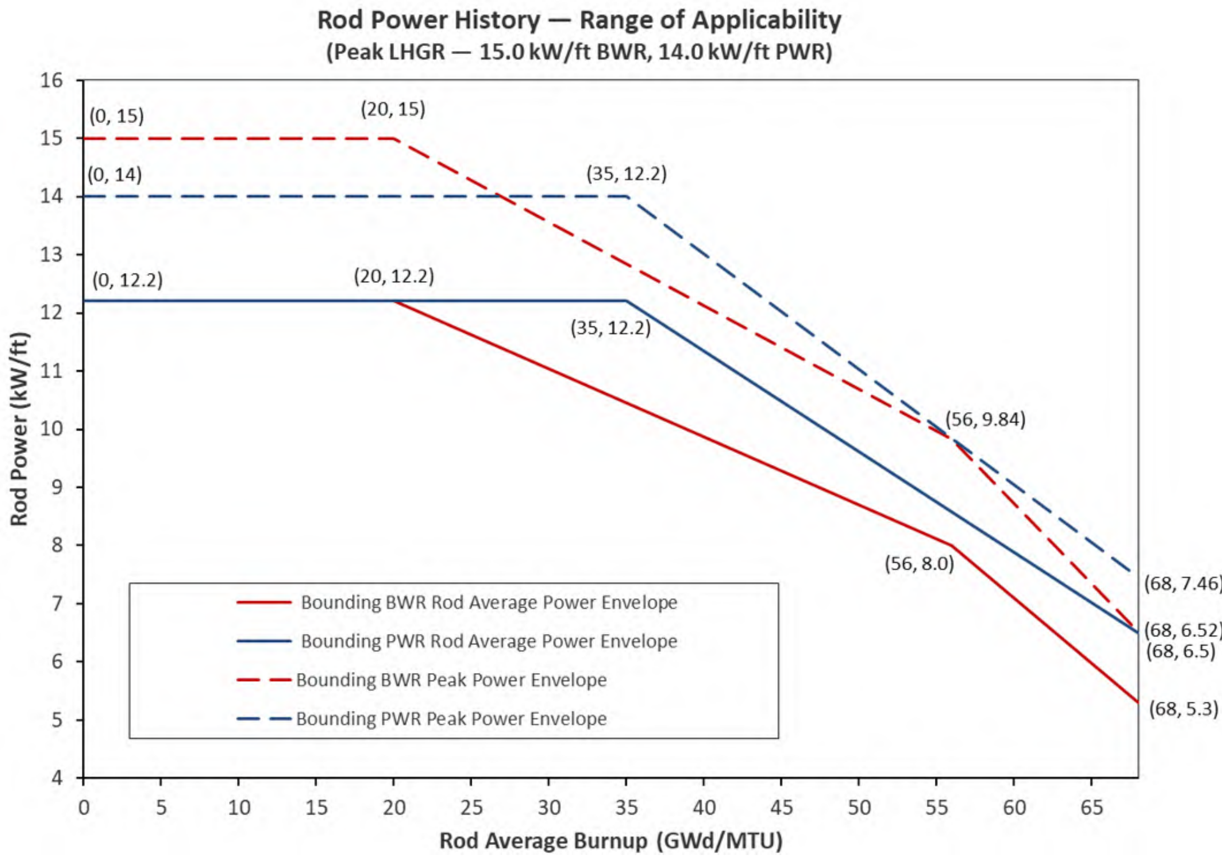

Figure 1. Maximum allowable power operating envelope for steady-state release fractions

For non-LOCA DBAs involving a rapid increase in fuel rod power, such as the BWR control rod drop accident and PWR control rod ejection accident, additional fission product releases may occur as a result of pellet fracturing and grain boundary separation. This transient fission gas release (T_{FGR}) increases the amount of activity available for release into the reactor coolant system for fuel rods that experience cladding breach. The empirical database suggests that T_{FGR} is sensitive to both local fuel burnup and peak radial average fuel enthalpy rise. As a result, separate low-burnup and high-burnup T_{FGR} correlations for stable, long-lived radionuclides (e.g., krypton (Kr)-85 and cesium-137) are provided, as follows:

pellet burnup < 50 GWd/MTU,

$$T_{FGR} \text{ (long-lived isotopes)} = \text{maximum } [\ (0.26 * \Delta H) - 13) / 100, 0\ ], \qquad \text{(Equation 1)}$$

pellet burnup ≥ 50 GWd/MTU,

$$T_{FGR} \text{ (long-lived isotopes)} = \text{maximum } [\ (0.26 * \Delta H) - 5) / 100, 0\ ], \qquad \text{(Equation 2)}$$

where

T_{FGR} = transient fission gas release fraction, and
ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

An investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release concluded that different radionuclides require adjustments to the above empirically based correlations (Ref. 28). For stable, long-lived noble gases (e.g., Kr-85) and alkali metals (e.g., cesium-137), the transient fission product release is equivalent to the above burnup-dependent correlations. For volatile, short-lived radioactive isotopes such as halogens (e.g., iodine (I)-131) and xenon (Xe) and Kr noble gases except Kr-85 (e.g., Xe-133, Kr-85m), the transient fission product release correlations should be multiplied by a factor of 0.333. The low-burnup and high-burnup T_{FGR} correlations for volatile, short-lived radioisotopes are as follows:

pellet burnup < 50 GWd/MTU,

T_{FGR} (short-lived isotopes) = 0.333 * maximum [(0.26 * ΔH) – 13) / 100, 0], (Equation 3)

pellet burnup ≥ 50 GWd/MTU,

T_{FGR} (short-lived isotopes) = 0.333 * maximum [(0.26 * ΔH) – 5) / 100, 0], (Equation 4)

where

T_{FGR} = transient fission gas release fraction, and
ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

For the remaining non-LOCA DBAs that predict fuel rod cladding failure, such as the PWR reactor coolant pump locked rotor and fuel handling accident, additional fission product releases may occur as a result of fuel pellet fragmentation (e.g., fracturing of high-burnup rim region) due to loss of pellet-to-cladding mechanical constraint or impact loads. T_{FGR} has been experimentally observed under a variety of accident conditions. At the time of issuance of Revision 1 of this RG, no consensus exists on the mechanism or the computation of T_{FGR} for these events; therefore, future applicants should address this using engineering judgment or experimental data. Though not fully applicable to non-LOCA and non-reactivity-initiated DBAs, NRC Research Information Letter 2021-13, "Interpretation of Research on Fuel Fragmentation Relocation, and Dispersal at High Burnup," issued December 2021 (Ref. 29), provides data that can be used to provide a bounding estimate of T_{FGR} for high-temperature DBAs.

The total fraction of fission products available for release equals the steady-state fission product release fractions in tables 3 and 4 plus any T_{FGR} prompted by the accident conditions. T_{FGR} may be calculated separately for each axial node based on local accident conditions (e.g., fuel enthalpy rise) and then combined to yield the total T_{FGR} for a particular damaged fuel rod. An NRC internal memorandum (Ref. 24) documents the technical bases of the steady-state fission product release fractions and T_{FGR} correlations.

The non-LOCA fission product release fractions and T_{FGR} correlations do not include the additional contribution associated with fuel melting. The event-specific appendices to this RG provide guidance for adjusting these gap inventories for fuel rods that are predicted to experience limited fuel centerline melting.

3.3 Timing of Release Phases

Table 5 provides the onset and end time of each sequential release phase for LOCA DBAs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel release phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[11] For

11 This statement excludes the effects of radioactive decay in the core inventory on the linear release modeled. In lieu of treating the release in a linear ramp manner, the activity for each phase can be modeled as being released instantaneously at the start of that release phase (i.e., in step increases).

non-LOCA DBAs in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage.

The applicability of table 5 is consistent with the applicability of tables 1 and 2.

Table 5. MHA LOCA Release Phases

Phase	PWRs		BWRs	
	Onset	End Time	Onset	End Time
Gap Release	0.5 minutes	0.23 hours	2 minutes	0.19 hours
Early In-Vessel	0.23 hours	4.5 hours	0.19 hours	8.0 hours

For facilities licensed with a leak-before-break methodology, the licensee may assume the onset of the gap release phase to be 10 minutes. The licensee may propose an alternative time for the onset of the gap release phase based on facility-specific calculations using suitable analysis codes, or based on an accepted topical report shown to apply to the specific facility. In the absence of approved alternatives, the licensee should use the gap release phase onsets in table 5.

3.4 Radionuclide Composition

Table 6 lists the elements in each radionuclide group that should be considered in design basis analyses.

Table 6. Radionuclide Groups

Group	Elements
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se
Barium, Strontium	Ba, Sr
Noble Metals	Ru, Rh, Pd, Co
Lanthanides	La, Nd, Eu, Pm, Pr, Sm, Y, Cm, Am
Cerium	Ce, Pu, Np, Zr
Molybdenum	Mo, Tc, Nb

3.5 Chemical Form

Of the radioiodine released from the reactor coolant system to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this RG contain additional details.

3.6 Fuel Damage in Non-Loss-of-Coolant-Accident Design Basis Accidents

The amount of fuel damage caused by non-LOCA DBAs should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel cladding is breached. Cladding failure mechanisms include high-temperature failure modes (e.g., critical heat flux, local oxidation, and ballooning) and pellet-to-cladding mechanical interaction.

Appendix B to this guide addresses the modeling of the amount of fuel damage caused by a fuel handling accident.

4. Dose Calculation Methodology

The NRC staff has determined (e.g., in “Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants: Final Rule” (61 FR 65157; December 11, 1996)) that there is an implied synergy between the ASTs and TEDE criteria and between the TID-14844 source terms and the whole-body and thyroid dose criteria. The TEDE criteria will not be used with results calculated from TID-14844. The guidance in this regulatory position applies to all dose calculations performed with an AST pursuant to 10 CFR 50.67 and 10 CFR Part 52. The regulatory position also provides guidance for determining control room and offsite doses and the control room and offsite dose acceptance criteria. Certain selective implementations may not require dose calculations, as described in Regulatory Position 1.3 of this guide.

4.1 Offsite Dose Consequences

The licensee should use the following assumptions in determining the TEDE for persons located at or beyond the EAB:

a. The dose calculations should determine the TEDE. TEDE is the sum of the effective dose equivalent (for external exposures) (EDEX) and the committed effective dose equivalent (for internal exposures) (CEDE). The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that have significant dose consequences and significant released radioactivity.[12]

b. The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data in International Commission on Radiological Protection (ICRP) Publication 30, “Limits for Intakes of Radionuclides by Workers,” issued in 1979 (Ref. 30). Table 2.1 of Federal Guidance Report 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” issued in 1988 (Ref. 31), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed “Effective” yield doses that correspond to the CEDE.

c. Table III.1 of Federal Guidance Report 12, “External Exposure to Radionuclides in Air, Water, and Soil,” issued in 1993 (Ref. 32), provides external effective dose equivalent (EDE) conversion factors acceptable to the NRC staff. The factors in the column headed “Effective” yield doses that correspond to the EDE.

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12 The prior practice of basing inhalation exposure on only radioiodine and not including radioiodine in external exposure calculations is not consistent with the definition of TEDE and the characteristics of the revised source term.

d. No correction should be made for depletion of the effluent plume by deposition on the ground.

e. The TEDE should be determined for an individual at the most limiting EAB location. The maximum EAB TEDE for any 2-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67[13] and 10 CFR Part 52. The maximum 2-hour TEDE should be determined by calculating the postulated dose for a series of small, time increments and performing a "sliding" sum over the increments for successive 2-hour periods. The maximum TEDE obtained is taken as the result of the analysis. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see analysis release duration in table 7). The analysis should assume that the most limiting 2-hour EAB χ/Q value occurs simultaneously with the limiting release to the environment (see also Regulatory Position 5.3 of this guide). In calculations of the maximum EAB TEDE for an individual, the maximum 2-hour EAB χ/Q value and a breathing rate of this individual of 3.5×10^{-4} cubic meters per second (m^3/s) should be used for the entire duration of the release to the environment to ensure that the limiting case is identified.

If multiple release paths are analyzed separately, additional processing is needed to identify the maximum 2-hour TEDE that is the sum of all paths, since the maximum periods may not be the same for each path. In these cases, it will be necessary to assess each release using the maximum 2-hour EAB χ/Q value, sum the doses for each pathway for each time increment, and then identify the maximum 2-hour EAB TEDE. As a conservative alternative, the maximum 2-hour TEDE for each path could be summed to determine the value for the accident.

f. The TEDE should be determined for the most limiting receptor at the outer boundary of the LPZ for the duration of the accident. This value should be used in determining compliance with the dose criteria in 10 CFR 50.67 and 10 CFR Part 52.

For the first 8 hours, the breathing rate of persons off site should be assumed to be 3.5×10^{-4} m^3/s. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} m^3/s. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} m^3/s.

13 For the EAB TEDE, the maximum 2-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the 2-hour window are considered only in the context of their impact on the maximum 2-hour EAB TEDE.

Table 7.[1] Accident Dose Criteria for EAB, LPZ, and Control Room Locations

Accident or Case	EAB and LPZ Dose Criteria (TEDE)	Control Room Dose Criteria[2] (TEDE)	Analysis Release Duration
MHA LOCA	0.25 sievert (Sv) (25 rem)	0.05 Sv (5.0 rem)	30 days for containment, ECCS, and MSIV (BWR) leakage
BWR Main Steamline Break			Instantaneous puff
Fuel Damage or Pre-Accident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Equilibrium Iodine Activity	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
BWR Rod Drop Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	24 hours
PWR Steam Generator Tube Rupture			Affected steam generator: time to isolate[3] Unaffected steam generator(s): until shutdown cooling is in operation and releases from the steam generator have been terminated
Fuel Damage or Pre-Accident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Concurrent Iodine Spike	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
PWR Main Steamline Break			Until shutdown cooling is in operation and releases from the steam generators have been terminated
Fuel Damage or Pre-Accident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Concurrent Iodine Spike	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
PWR Locked Rotor Accident	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	Until shutdown cooling is in operation and releases from the steam generators have been terminated
PWR Control Rod Ejection Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	Containment pathway: 30 days; Secondary system: until shutdown cooling is in operation and releases from the steam generators have been terminated
Fuel Handling Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	30 days

[1] For PWRs with steam generator alternative repair criteria, different dose criteria may apply to steam generator tube rupture and main steamline break analyses.

[2] The control room exposure period is 30 days for all accidents.

[3] Tube rupture in the affected steam generator may result in the need to control steam generator water level using steam dumps. These releases may extend the duration of the release from the affected steam generator beyond the initial isolation.

The column labeled “Analysis Release Duration” summarizes the assumed radioactivity release durations identified in the individual appendices to this guide. These appendices contain complete descriptions of the release pathways and durations.

4.2 Control Room Dose Consequences

The following guidance should be used in determining the TEDE for persons located in the control room.

4.2.1 Sources of Radiation

The TEDE analysis should consider all sources of radiation that will cause exposure of control room personnel. The applicable sources will vary from facility to facility but typically will include the following:

a. contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,

b. contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope,

c. radiation shine from the external radioactive plume released from the facility,

d. radiation shine from radioactive material in the reactor containment, and

e. radiation shine from radioactive material in systems and components inside or external to the control room envelope (e.g., radioactive material buildup in recirculation filters).

4.2.2 Material Releases and Radiation Levels

The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, in-plant transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would produce nonconservative results for the control room.

4.2.3 Transport Models

The models used for the transport of radioactive material into and through the control room[14] and the shielding models used to determine radiation dose rates from external sources should be structured to provide suitably conservative estimates of the exposure of control room personnel.

4.2.4 Engineered Safety Features

The licensee may assume credit for ESFs that mitigate airborne radioactive material within the control room. Such features may include control room isolation or pressurization or intake or recirculation filtration. Guidance appears in SRP Section 6.5.1, “ESF Atmosphere Cleanup Systems,” and RG 1.52, Revision 4, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of

14 The iodine protection factor methodology of Reference 30 may not be adequately conservative for all DBAs and control room arrangements because it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the iodine protection factor methodology should be used only with caution. The NRC computer codes HABIT (Ref. 33) and RADTRAD (Ref. 34) incorporate suitable methodologies.

Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," issued September 2012 (Ref. 35). The control room design is often optimized for the MHA LOCA, and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by ESF signals or radiation monitors. In some cases, the ESF signal is effective only for selected accidents, placing reliance on the radiation monitors for the remaining accidents. Several aspects of radiation monitors can delay control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

4.2.5 Personal Protective Equipment

The licensee should generally not take credit for the use of personal protective equipment or prophylactic drugs such as potassium iodide. The NRC may consider deviations on a case-by-case basis.

4.2.6 Dose Receptor

The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100 percent of the time during the first 24 hours after the event, 60 percent of the time between 1 and 4 days, and 40 percent of the time from 4 days to 30 days.[15] For the duration of the event, the licensee should assume the breathing rate of this individual to be 3.5×10^{-4} m^3/s (Ref. 38).

4.2.7 Dose Conversion Factor

The licensee should calculate control room doses using the dose conversion factors identified in Regulatory Position 4.1 for use in offsite dose analyses. The calculation should consider all radionuclides, including progeny from the decay of parent radionuclides that have significant dose consequences, and the released radioactivity. The EDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, $EDEX_{\infty}$, to a finite cloud dose, $EDEX_{finite}$, where the control room is modeled as a hemisphere that has a volume, *V*, in cubic feet, equivalent to that of the control room (Ref. 36):

$$EDEX_{finite} = \frac{EDEX_{\infty} V^{0.338}}{1173} \qquad \text{(Equation 5)}$$

The expression may be already incorporated into certain radiological assessment codes and would not need to be separately added to the dose results of finite volumes such as control room doses when using those codes.

4.3 Other Dose Consequences

The licensee should use the guidance in Regulatory Positions 4.1 and 4.2, as applicable, to reassess the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737. The licensee should update design envelope source terms from NUREG-0737 for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be

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15 These occupancy factors are already included in the determination of the χ/Q values using the Murphy and Campe methodology described in "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," issued August 1974 (Ref. 36), and should not be credited twice. The ARCON96 code (Ref. 37) does not incorporate these occupancy factors into the determination of the χ/Q values. Therefore, dose calculations using ARCON96 χ/Q values should include the occupancy factors.

expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance in RG 1.89, Revision 2.

4.4 Acceptance Criteria

The accident dose radiological criteria for the EAB, for the outer boundary of the LPZ, and for the control room are in 10 CFR 50.34, 10 CFR Part 52, 10 CFR 50.67, and GDC 19 in Appendix A to 10 CFR Part 50. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation. For events with a higher probability of occurrence (e.g., fuel handling accidents), postulated EAB and LPZ doses should not exceed the criteria in table 7. The accident dose for the EAB should not exceed the acceptance criteria for any 2-hour period following the onset of the fission product release. The accident dose for the LPZ should not exceed the acceptance criteria during the entire period of the passage of the fission product release.

The acceptance criteria for the various NUREG-0737 items generally reference GDC 19 or specify criteria derived from GDC 19. These criteria are generally specified in terms of whole-body dose or its equivalent to any body organ. For facilities applying for, or having received, approval to use an AST, licensees should update the applicable criteria for consistency with the TEDE criterion in 10 CFR 50.67(b)(2)(iii).

For new reactor applicants, the technical support center (TSC) habitability acceptance criterion is based on the requirement of paragraph IV.E.8 of Appendix E to 10 CFR Part 50 to provide an onsite TSC from which effective direction can be given and effective control can be exercised during an emergency. The radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria specified for the control room (5 rem TEDE for the duration of the accident).

5. Analysis Assumptions and Methodology

5.1 General Considerations

5.1.1 Analysis Quality

The analyses discussed in this guide are reanalyses of the design basis safety analyses required by 10 CFR 50.67 or evaluations required by 10 CFR 50.34, 10 CFR Part 52, and GDC 19. These analyses are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59 and 10 CFR Part 52. The licensee should prepare, review, and maintain these analyses in accordance with quality assurance programs that comply with Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50.

These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative bounding assumptions rather than being modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based on data from specific accident sequences, since the DBAs were never intended to represent any specific accident sequence; the proposed deviation may not be conservative for other accident sequences.

5.1.2 Credit for Engineered Safeguard Features

The licensee may take credit for accident mitigation features that are classified as safety related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. However, the licensee should not take credit for ESFs that would affect the generation of the source term described in tables 1 and 2. Additionally, the licensee should assume the single active component failure that results in the most limiting radiological consequences. Assumptions about the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences. The licensee should consider design basis delays in the actuation of these features, especially for features that rely on manual intervention.

5.1.3 Assignment of Numerical Input Values

The licensee should select the numerical values to be used as inputs to the dose analyses with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but nonconservative in another portion of the same analysis. For example, an assumption of minimum containment system spray flow is usually conservative for estimating iodine scrubbing but, in many cases, may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a given parameter for the duration of the event, particularly for parameters affected by changes in density. For a parameter addressed by technical specifications, the value used in the analysis should be that identified in the technical specifications.[16] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing (e.g., steam generator nondestructive testing), the degradation that may occur between periodic tests should be considered in establishing the analysis value.

5.1.4 Applicability of Prior Licensing Basis

The NRC staff considers the implementation of an AST to be a significant change to the design basis of the facility that is voluntarily initiated by the licensee. The characteristics of the ASTs and the revised dose calculation methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. Licensees should consider and address new or unreviewed issues created by a particular site-specific implementation of the AST where the implementation conflicts with the facility's licensing basis. However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as part of the facility's design basis. Licensees should ensure that analysis assumptions and methods are compatible with the ASTs and the TEDE criteria.

5.2 Accident-Specific Assumptions

The appendices to this RG provide accident-specific assumptions that are acceptable to the staff for performing site-specific analyses as required by 10 CFR 50.34, 10 CFR Part 52, 10 CFR 50.67, and GDC 19. Licensees should review their licensing basis documents for guidance on the analysis of radiological DBAs other than those provided in this guide. The DBAs addressed in these attachments

16 Note that for some parameters, the technical specification value may be adjusted, for analysis purposes, by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in RG 1.52, rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address possible changes in the parameter between scheduled surveillance tests.

were selected from accidents that may involve damage to irradiated fuel. This guide does not address all DBAs with radiological consequences. The inclusion or exclusion of a particular DBA in this guide does not mean that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST and changes to the facility or to the radiological analyses.

The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses, and the NRC staff generally expects licensees to address each assumption or to propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses, or, in some cases, a previously approved licensing-basis consideration. The assumptions in the appendices are consistent with the AST identified in Regulatory Position 3. Although applicants are free to propose alternatives to these assumptions for consideration by the NRC staff, the use of staff positions inconsistent with these assumptions is beyond the scope of this guidance.

5.3 Atmospheric Dispersion Modeling and Meteorology Assumptions

Atmospheric dispersion factors (χ/Q values) for the EAB, the LPZ, the control room, and, as applicable, the onsite emergency response facility (i.e., the TSC)[17] that the staff approved during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified in this guide, provided that such values remain relevant to the particular accident, release characteristics that affect plume rise, its release points, and receptor locations. Licensees should ensure that any previously approved values remain accurate and do not include any misapplication of a methodology or calculational errors in the identified values. RG 1.145, Revision 1, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” issued November 1982 (Ref. 39), and RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” issued June 2003 (Ref. 40), document methods for determining χ/Q values.

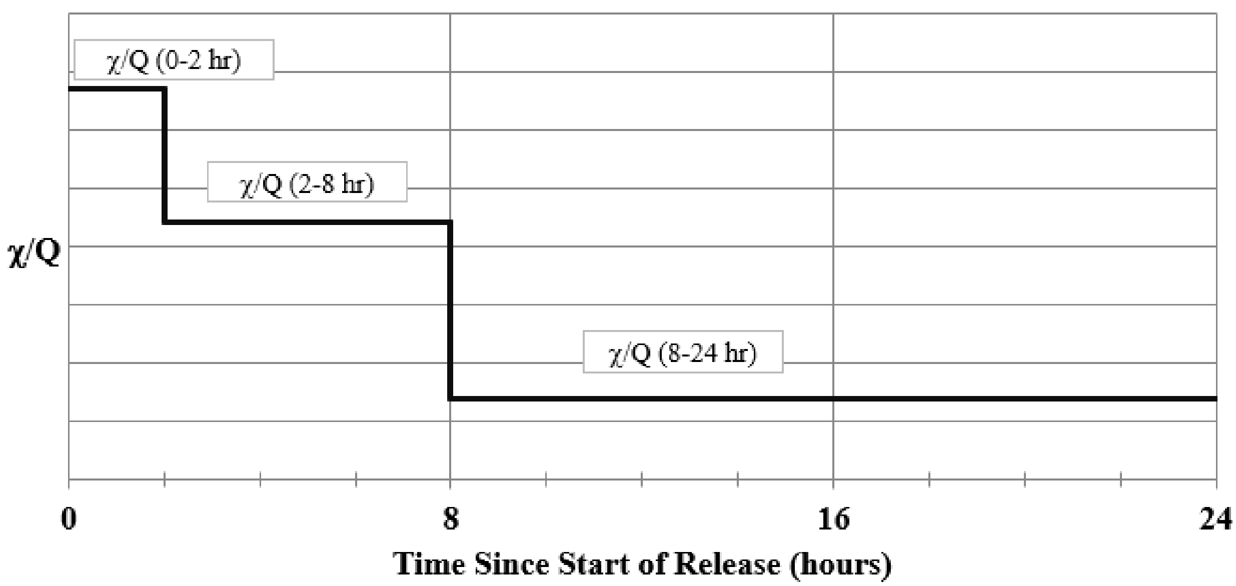
RG 1.145 and RG 1.194 should be used if the FSAR χ/Q values are to be revised, or if values are to be determined for new release points, receptor distances, or release characteristics that affect plume rise. In addition to calculating control room χ/Q values, the modeling methodology outlined in RG 1.194 may be modified to estimate offsite χ/Q values at offsite boundaries out to distances of 1,200 m if using the procedures consistent with RG 1.249, “Use of ARCON Methodology For Calculation Of Accident-Related Offsite Atmospheric Dispersion Factors” (Ref. 41). EAB χ/Q values are determined for the limiting 2-hour period within a 30-day period following the start of the radioactivity release. Control room χ/Q values are generally determined for initial averaging periods of 0–2 hours and 2–8 hours, and LPZ χ/Q values are generally determined for an initial averaging period of 0–8 hours. Control room and LPZ χ/Q values are also generally determined for averaging periods of 8–24 hours, 24–96 hours, and 96–720 hours.

The source term defined in TID-14844 assumes that the entire source term is instantaneously released into the containment atmosphere. Therefore, the maximum release rate coincides with the most conservative 0–2 hour χ/Q value. In contrast, the AST is assumed to develop over specified time intervals, with the maximum release rate occurring sometime after accident initiation. To ensure a conservative dose analysis, the period of the most adverse release of radioactive materials to the environment, with respect to doses, should be assumed to occur coincident with the period of most unfavorable atmospheric

17 The radiological habitability analysis for an onsite TSC is performed to support the emergency preparedness review of emergency facilities. Reevaluation of TSC habitability as part of a license amendment request may be needed if a radiological analysis of the TSC is included in that plant’s current licensing basis. For an onsite TSC, the atmospheric dispersion modeling is handled similarly to that for the control room.

dispersion. One acceptable methodology for calculating the control room and LPZ χ/Q values is as follows. If the 0–2 hour χ/Q value is calculated, this value should be used coincident with the maximum 2-hour release to the environment. If the maximum 2-hour release occurs at the beginning of the period of releases to the environment, the 2–8 hour χ/Q value should be used for the remaining 6 hours of the first 8-hour time period. If the maximum 2-hour release occurs sometime after the beginning of the releases, the 2–8 hour χ/Q value should be used before and after the maximum 2-hour release for a combined total of 6 hours. The 8–24, 24–96, and 96–720 hour χ/Q values should similarly be used for the remainder of the release duration. Figure 2 provides examples of aligning χ/Q values with the maximum 2-hour release.

a. Maximum 2-hour release occurs during hours 0–2.



b. Maximum 2-hour release occurs during hours 6–8.

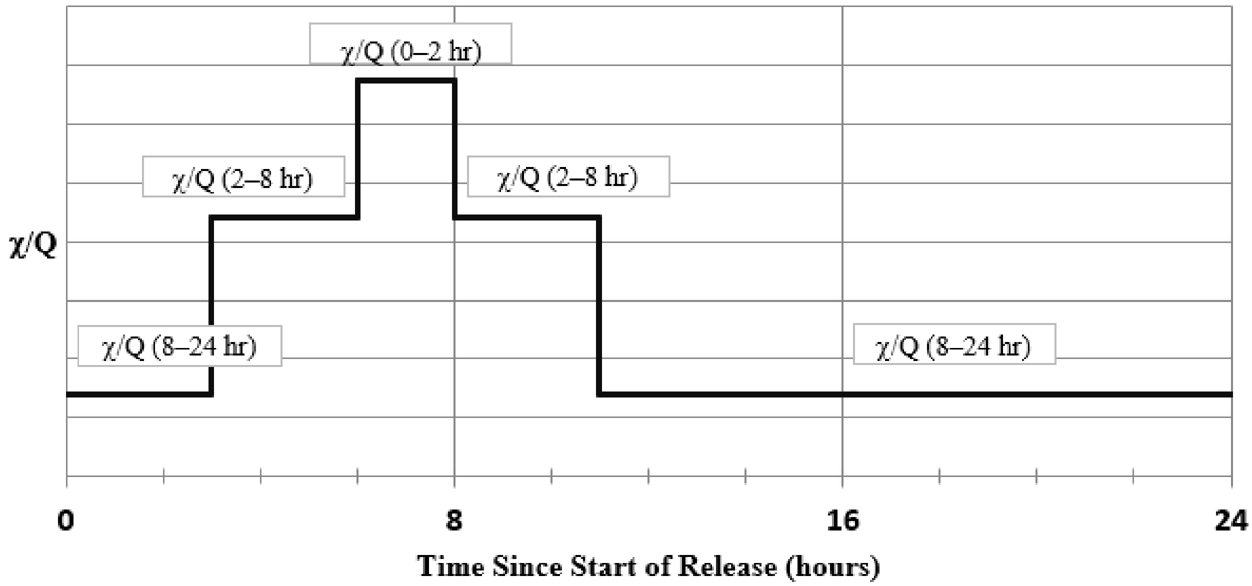


Figure 2. Example alignments of χ/Q values with the maximum 2-hour release period

D. IMPLEMENTATION

The NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, "Backfitting," and as described in NRC Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 42), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

REFERENCES[1]

1. Sandia National Laboratories, SAND2011-0128, “Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel,” Albuquerque, New Mexico, January 2011 (ML20093F003).

2. U.S. Nuclear Regulatory Commission (NRC), RG 1.183, Revision 0, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Washington, DC, July 2000 (ML003716792).

3. *U.S. Code of Federal Regulations* (CFR), “Domestic Licensing of Production and Utilization Facilities,” Part 50, Chapter I, Title 10, “Energy.”

4. CFR, “Licenses, Certifications, and Approvals of Nuclear Power Plants,” Part 52, Chapter I, Title 10, “Energy.”

5. CFR, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” Part 54, Chapter I, Title 10, “Energy.”

6. NRC, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Washington, DC.

7. American National Standards Institute (ANSI)/American Nuclear Society (ANS), ANSI/ANS-5.4, “Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel,” La Grange Park, Illinois, May 2011 (not publicly available in ADAMS).

8. NRC, “Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement,” *Federal Register*, Vol. 60, pp. 42622–42629 (60 FR 42622), Washington, DC, August 16, 1995.

9. NRC, NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” Washington, DC, February 1995 (ML041040063).

10. U.S. Atomic Energy Commission (now U.S. Nuclear Regulatory Commission), TID-14844, “Calculation of Distance Factors for Power and Test Reactor Sites,” Washington, DC, March 1962 (ML021720780).

11. CFR, “Reactor Site Criteria,” Part 100, Chapter I, Title 10, “Energy.”

12. NRC, RG 1.89, Revision 2, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” Washington, DC, April 2023 (ML22272A602).

1 Publicly available NRC published documents are available electronically through the NRC Library on the NRC’s public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC’s Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

13. NRC, "Nuclear Regulatory Commission International Policy Statement," *Federal Register*, Vol. 79, No. 132, pp. 39415–39418 (79 FR 39415), Washington, DC, July 10, 2014.

14. NRC, Management Directive 6.6, "Regulatory Guides," Washington, DC, July 19, 2022 (ML22010A233).

15. NRC, NUREG-0396, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," Washington, DC, December 1978 (ML051390356).

16. NRC, NUREG-0654, Revision 1 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Washington, DC, November 1980 (ML040420012).

17. CFR, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Part 51, Chapter I, Title 10, "Energy."

18. NRC, NUREG-0737, "Clarification of TMI Action Plan Requirements," Washington, DC, November 1980 (ML102560051).

19. NRC, SECY-98-154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," Washington, DC, June 30, 1998 (ML992880064).

20. NRC, RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Washington, DC, January 2018 (ML17317A256).

21. NRC, Office of Nuclear Reactor Regulation Office Instruction LIC-206, Revision 1, "Integrated Risk-Informed Decision-Making for Licensing Reviews," Washington, DC, June 26, 2020 (ML19263A645).

22. NRC, RG 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Washington, DC, November 1978 (ML011340122).

23. NRC, RG 1.206, Revision 1, "Applications for Nuclear Power Plants," Washington, DC, October 2018 (ML18131A181).

24. Clifford, P., NRC, internal memorandum to R. Lukes, "Technical Basis for Draft RG 1.183 Revision 1 (2021) Non-LOCA Fission Product Release Fractions," Washington, DC, July 28, 2021 (ML21209A524).

25. Case, M., NRC, internal memorandum to J. Donoghue and M. Franovich, "Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment," Washington, DC, May 13, 2020 (ML20126G376).

26. NRC, NUREG/CR-7282, "Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases," Washington, DC, July 2021 (ML21210A321).

27. Esmaili, H., NRC, internal memorandum to K. Hsueh, "Letter Report on Evaluation of the Impact of Fuel Fragmentation, Relocation, and Dispersal for the Radiological Design Basis Accidents in Regulatory Guide 1.183," Washington, DC, July 20, 2021 (ML21197A067).

28. Pacific Northwest National Laboratory, Report 18212, Revision 1, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," Richland, Washington, June 2011 (ML112070118).

29. NRC, Research Information Letter 2021-13, "Interpretation of Research on Fuel Fragmentation Relocation, and Dispersal at High Burnup," Washington, DC, December 2021 (ML21313A145).

30. International Commission on Radiological Protection (ICRP), ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers," Pergamon Press, London, 1979.[2]

31. U.S. Environmental Protection Agency (EPA), EPA-520/1-88-020, Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Washington, DC, 1988.[3]

32. EPA, EPA-402-R-93-081, Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil," Washington, DC, 1993.

33. NRC, NUREG/CR-6210, Supplement 1, "Computer Codes for Evaluation of Control Room Habitability (HABIT V1.1)," Washington, DC, October 1998 (not publicly available in ADAMS).

34. NRC, NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," Washington, DC, April 1998 (ML15092A284); NUREG/CR-7220, "SNAP/RADTRAD 4.0: Description of Models and Methods," Washington, DC, June 2016 (ML16160A019).

35. NRC, RG 1.52, Revision 4, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Washington, DC, September 2012 (ML12159A013).

36. Murphy, K.G., and K.W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," Proceedings of 13th AEC Air Cleaning Conference, U.S. Atomic Energy Commission (now U.S. Nuclear Regulatory Commission), Washington, DC, August 1974 (ML19116A064).

37. NRC, NUREG/CR-6331, Revision 1, "Atmospheric Relative Concentrations in Building Wakes," Washington, DC, May 1997 (ML17213A190).

38. ICRP, ICRP Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation," Pergamon Press, London, 1959.

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2 Copies of International Commission on Radiological Protection (ICRP) documents may be obtained through the ICRP website: http://www.icrp.org/; 280 Slater Street, Ottawa, Ontario K1P 5S9, Canada; tel: +1(613) 947-9750, fax: +1(613) 944-1920.

3 Copies of EPA Library Services documents may be obtained through the agency's website: https://www.epa.gov/libraries.

39. NRC, RG 1.145, Revision 1, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Washington, DC, November 1982 (ML003740205).

40. NRC, RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Washington, DC, June 2003 (ML031530505).

41. NRC, RG 1.249, "Use of ARCON Methodology For Calculation Of Accident-Related Offsite Atmospheric Dispersion Factors," Washington, DC, August 2023 (ML22024A241).

42. NRC, Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Washington, DC, September 2019 (ML18093B087).

APPENDIX A

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF LIGHT-WATER REACTOR MAXIMUM HYPOTHETICAL LOSS-OF-COOLANT ACCIDENTS

The assumptions in this appendix are acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of maximum hypothetical accident (MHA) loss-of-coolant accidents (LOCAs) at light-water reactors. These assumptions supplement the guidance in the main body of this guide.

Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. A-1), defines LOCAs as those postulated accidents that result from a loss-of-coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system (RCS) are included. The MHA LOCA, like all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge aspects of the facility design. Separate mechanistic analyses are performed using a spectrum of break sizes to evaluate fuel and emergency core cooling system performance for conformance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." With regard to radiological consequences, an MHA LOCA is typically assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility. The analysis should calculate the limiting dose consequences to the public and control room doses, assuming a deterministic substantial core damage source term, discussed below, released into an intact containment.

A-1. Source Term

Regulatory Position 3 of this guide provides acceptable assumptions about core inventory and the release of radionuclides from the fuel.

A-1.1 If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the MHA LOCA event (e.g., radiolysis products). With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

A-2. Transport in Primary Containment

Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in a pressurized-water reactor (PWR) or the drywell in a boiling-water reactor (BWR) are as follows:

A-2.1 The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment (in a PWR) or the drywell (in a BWR) as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included, provided there is a mechanism to ensure mixing between the drywell and the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel release phase.

A-2.2 Reduction in airborne radioactivity in the containment due to natural deposition within the containment may be credited. Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Ref. A-2), describes an acceptable model for removal of iodine and aerosols. The NRC staff no longer accepts the prior practice of deterministically assuming that a 50 percent plateout of iodine is released from the fuel, or the aerosol reductions (decontamination) calculated in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," issued July 1996 (Ref. A-3), because the deterministic 50 percent plateout value and the aerosol reductions calculated in NUREG/CR-6189 are inconsistent with the characteristics of the revised source terms. However, the methods used in NUREG/CR-6189 may be credited on a case-by-case basis if they are adjusted to incorporate the revised MHA LOCA source term in this revision of Regulatory Guide (RG) 1.183. When these adjusted NUREG/CR-6189 methods are used, the DBA analyses should use the 10th-percentile values unless otherwise justified. Some licensees may consider specific containment design features to evaluate aerosol fission product removal. The amount of removal will be evaluated on a case-by-case basis. Reduction in airborne aerosol radioactivity in the containment by both sprays and gravitational settling should be evaluated on a case-by-case basis.

A-2.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with section 6.5.2 of the SRP (Ref. A-2) may be credited. Section 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," issued June 1993 (Ref. A-4), describe acceptable models for the removal of iodine and aerosols (DBA analyses should use the 10th-percentile values). The analysis code RADionuclide, Transport, Removal and Dose Estimation (RADTRAD) (Ref. A-5) incorporates this simplified model.

The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray droplets. In addition, since spray droplets are assumed to be ineffective once they impact a structure, the obstructions in drywells and containments (particularly in BWR Mark I and Mark II drywells) should be considered in the determination of decontamination factors (DFs) and removal coefficients credited for the drywell or containment. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed region volume per hour, unless other rates are justified. On a case-by-case basis, the licensee may consider containment mixing rates determined by the cooldown rate in the sprayed region and the buoyancy-driven flow that results. The containment building atmosphere may be considered a single well-mixed volume if the spray covers at least 90 percent of the containment building space and an engineered-safety-feature (ESF) ventilation system is available for adequate mixing of the unsprayed compartments.

As provided in the SRP, the maximum DF for elemental iodine is based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in tables 1 and 2 of this guide, multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., the SRP methodology treats aerosols as particulate).

A-2.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of RG 1.52, Revision 4, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," issued September 2012 (Ref. A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.

A-2.5 Historically, reduction in airborne radioactivity in the containment by suppression pool scrubbing has not been credited in licensing actions for operating BWRs; however, the staff may consider such reduction on a case-by-case basis.[1] The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. A-8). For suppression pool solutions having a pH less than 7, elemental iodine vapor should be conservatively assumed to evolve into the containment atmosphere.

A-2.6 Reduction in airborne radioactivity in the containment by retention in ice condensers, or other ESFs not addressed above, should be evaluated on a case-by-case basis. See SRP Section 6.5.4, "Ice Condenser as a Fission Product Cleanup System" (Ref. A-2).

A-2.7 The evaluation should assume that the primary containment (i.e., drywell and wetwell for Mark I and II containment designs) will leak at the peak pressure technical specification (TS) leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50 percent of the TS leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50 percent of the TS leak rate. Leakage from sub-atmospheric containments is assumed to terminate when the containment is brought to and maintained at a sub-atmospheric condition as defined by the TS.

A-2.8 If the primary containment is routinely purged during power operations, the licensee should analyze releases via the purge system before containment isolation and should sum the resulting doses with the postulated doses from other release paths. The purge release evaluation should assume that 100 percent of the radionuclide inventory in the RCS liquid is released to the containment at the initiation of the MHA LOCA. This inventory should be based on the TS RCS equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the licensee should consider release fractions associated with the gap release and early in-vessel release phases as applicable.

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1 For an example of the modeling of radionuclide transport in containment with scrubbing credit in the primary containment cooling system of a new BWR reactor application, see section 15.4.5 of NUREG-1966, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design," issued April 2014 (ML14100A304 (package)) (Ref. A-7).

A-3. Dual Containments

For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows:

A-3.1 Leakage from the primary containment should be considered to be collected, processed by ESF filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in the TS. Credit for an elevated release should be assumed only if the point of physical release is more than 2.5 times the height of any adjacent structure.

A-3.2 Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in the TS.

A-3.3 The effect of high windspeeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on a case-by-case basis. The windspeed to be assumed is the 1-hour average value that is exceeded only 5 percent of the total number of hours in the dataset. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded in either 5 percent or 95 percent of the total numbers of hours in the dataset, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5 percent of the time) (Ref. A-9).

A-3.4 Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50 percent. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede streamflow between the release and the exhaust.

A-3.5 Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the TS. If the bypass leakage is through water (e.g., via a filled piping run that is maintained full), credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosols and elemental halogens in gas-filled lines may be considered on a case-by-case basis.

A-3.6 Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account, provided that these systems meet the guidance of RG 1.52 (Ref. A-6).

A-4. Assumptions on Engineered-Safety-Feature System Leakage

ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-8). The licensee should analyze the radiological consequences from the postulated leakage and combine them with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the MHA LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs:

A-4.1 With the exception of noble gases, all fission products released from the fuel to the containment (as defined in tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameter values that make spray and deposition models conservative in estimating containment airborne leakage are nonconservative in estimating the buildup of sump activity.

A-4.2 The leakage should be taken as 2 times[2] the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the TS, or licensee commitments to item III.D.1.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980 (Ref. A-10), would require declaring such systems inoperable. Design leakage from any systems not included in the TS that transport primary coolant sources outside of containment should be added to the total leakage. The applicant should justify the design leakage used. The leakage should be assumed to start at the earliest time when the recirculation flow occurs in these systems, and to end at the latest time when the releases from these systems are terminated. It should account for the ESF leakage at accident conditions. Design leakage through valves isolating ESF recirculation systems from tanks vented to the atmosphere (e.g., the pump miniflow return to the refueling water storage tank in the emergency core cooling system) should also be considered.

A-4.3 With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.

A-4.4 If the temperature of the leakage exceeds 212 degrees Fahrenheit (°F), the fraction of total iodine (i.e., aerosol, elemental, and organic) in the liquid that becomes airborne should be assumed to equal the fraction of the leakage that flashes to vapor. This flash fraction (FF) should be determined using a constant enthalpy, *h*, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment, using the following formula:

$$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$

where

h_{f1} is the enthalpy of liquid at system design temperature and pressure,
h_{f2} is the enthalpy of liquid at saturation conditions (14.7 pounds per square inch absolute, 212°F), and
h_{fg} is the heat of vaporization at 212°F.

A-4.5 If the temperature of the leakage is less than 212°F or the calculated FF is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine

2 The multiplier of 2 is used to account for increased leakage in these systems over the duration of the accident and between surveillances or leakage checks.

activity in the leaked fluid, unless a smaller amount can be substantiated. The justification of such values should consider the sump pH history; changes to the leakage pH caused by pooling on concrete surfaces, leaching through piping insulation, evaporation to dryness, and mixing with other liquids in drainage sumps; area ventilation rates and temperatures; and subsequent re-evolution of iodine.

A-4.6 The radioiodine that is postulated to be available for release to the environment is assumed to be 97 percent elemental and 3 percent organic.[3] Reduction in release activity by dilution or hold-up within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated using the guidance of RG 1.52 (Ref. A-6).

A-5. Main Steam Isolation Valve Leakage in Boiling-Water Reactors

For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The licensee should analyze and combine the radiological consequences from postulated MSIV leakage with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the MHA LOCA.

Three methods are presented below to compute aerosol deposition within main steamlines. Each method computes similar removal coefficients that are suitable for radiological consequence calculations. These methods are not valid if credit is also taken for aerosol removal from drywell sprays, or for other containment aerosol removal processes, when modeling the MSIV leakage release pathway without accounting for the change in particle size distribution due to these containment removal processes. The three MSIV leakage models are the following:

a. direct adoption of the recommendations in SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," issued October 2008 (Ref. A-12), without scaling "R*" or "R_M" factors,

b. re-evaluated Accident Evaluation Branch (AEB)-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," dated December 9, 1998 (Ref. A-13), with multi-group, and

c. numerical integration.

The assumptions in the following subsections are acceptable for evaluating the consequences of MSIV leakage.

A-5.1 The source of the MSIV leakage should be assumed to be the containment (or drywell)[4] activity concentration (see Regulatory Position A-2.1).

For new BWR designs or license amendments that propose changes from a referenced design control document, other models of MSIV source concentration will be considered on a

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3 The 97 percent elemental, 3 percent organic speciation is a conservative deterministic assumption based on the hypothesis that most of the iodine released to the environment will be in elemental form, with a small percentage converted to organic, as supported in section 3.5 of NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," issued February 1995 (Ref. A-11).

4 Note that for the purpose of this analysis, the containment now extends up to the MSIVs, which are designated as containment isolation valves.

case-by-case basis. In general, the concepts used in developing the guidance for BWR Mark I, II, and III plants may be followed as applicable to designs under consideration.

A-5.2 The chemical form of radioiodine released to the drywell should be assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

A-5.3 All the MSIVs should be assumed to leak at the maximum leak rate above which the TS would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident, as specified in table 7 of this guide, and should be assigned to steamlines so that the accident dose is maximized. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50 percent of the maximum leak rate. Section 5.4 of SAND2008-6601 (Ref. A-12) describes an acceptable model for estimating the volumetric flow rate in the steamline.

A-5.4 A reduction in MSIV releases caused by hold-up and deposition in the main steam piping and main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake and are powered by emergency power sources. These reductions are allowed for steam system piping segments that are enclosed by physical barriers, such as closed valves. The piping segments and physical barriers should be designed, constructed, and maintained to seismic Category I guidelines as specified in RG 1.29, "Seismic Design Classification for Nuclear Power Plants" (Ref. A-14). Alternatively, operating license holders may evaluate and demonstrate the piping segments and barriers to be rugged as described in Regulatory Position A-5.5. The amount of reduction allowed will be evaluated on a case-by-case basis and is to be justified based on the alternative drain pathways established by operating procedures and the potential leakage pathways to the environment.

On March 3, 1999, the NRC staff issued a safety evaluation (Ref. A-15) of the GE topical report NEDC-31858P, Revision 2, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," issued September 1993 (Ref. A-16). In its safety evaluation, the staff found the Boiling Water Reactor Owners' Group (BWROG) report to be an acceptable method for direct reference in individual submittals on MSIV leakage, subject to the conditions and limitations described in the safety evaluation. For the purposes of DBA radiological consequence analyses, based on the information in the BWROG report, proposed MSIV leakage limits in excess of 200 standard cubic feet per hour (scfh) per steamline and in excess of 400 scfh for total MSIV leakage will be considered on a case-by-case basis with sufficient justification. The single valve limitation is based on the consideration that leakage in excess of 200 scfh may indicate a substantial valve defect. The total MSIV leakage limitation of 400 scfh is based on considerations of the relationship of MSIV leakage rate to the allowable containment leakage rate (L_a), as well as providing defense in depth related to the single valve limitation.

Consistent with the BWROG report, the following information related to the reliability of the pathway to the main condenser should be provided when an alternative drain pathway is credited:

a. the alternative drain pathway and the basis for its functional reliability, commensurate with its intended safety-related function;

b. the maintenance and testing program for the active components (such as valves) in the alternative drain pathway, and a confirmation that the valves that are required to open the alternative drain pathway are included in the inservice testing program;

c. how the alternative drain pathway addresses the single failure of active components to verify its availability to convey MSIV leakage to the condenser;

d. a secondary pathway to the condenser; and

e. emergency operating procedures that may be required to identify necessary operator actions to mitigate MSIV leakage consequences utilizing the alternative drain pathway if a highly reliable power source is available or to identify necessary operator actions to mitigate MSIV leakage consequences using the alternative drain pathway if a highly reliable power source is unavailable.

A-5.5 Licensees that have already evaluated the seismic ruggedness of the steamlines, alternate drain paths, and the main condenser, and who have obtained prior staff approval, may credit the piping addressed in that approval. Licensees that have not previously applied for such approval may do so in accordance with the guidance in Reference A-15, or using the revised seismic analysis method described below in "Revised Seismic Analysis of the Alternative Drain Pathway" for establishing a seismically rugged alternative drain pathway to the condenser.

Licensees choosing either of these methods should define the alternative drain pathway to the main condenser and provide the basis for its reliability. The basis for reliability should include the qualification and redundancy of valves that must change position to establish the pathway, operator training, and procedures governing establishment of the alternative drain pathway as described in Regulatory Position A-5.4.

The revised seismic analysis method described below in "Revised Seismic Analysis of the Alternative Drain Pathway" provides a method to demonstrate that the defined drain path to the condenser is seismically rugged. The NRC staff's determination of reasonable assurance that structures, systems, and components (SSCs) in the pathway will not undergo gross seismic failure is based upon the staff's consideration of engineering information, operating experience, and probabilistic insights related to seismic events, including the following:

- margin in material strength provided by use of an appropriate code of record for design of the pathway(s) to the condenser, and by construction to augmented quality standards in the areas of material certification, testing, and nondestructive examination;

- additional margin in material strength provided by the high-pressure and high-temperature design of the SSCs in the drain pathway to accommodate seismic loads after an accident considered for the analysis in this RG;

- low failure probability (5 percent or less) for the SSCs in the drain pathway at safe-shutdown earthquake seismic accelerations for the majority of operating plants, especially BWRs (ranging from 0.12g to 0.25g peak ground acceleration), based on a conservative median fragility value to represent these SSCs;

- available information from post-earthquake walkdowns performed for the nuclear power plants at Kashiwazaki-Kariwa in Japan and at North Anna Power Station in the United States, and consideration of the walkdowns performed by licensees to identify weaknesses in SSCs when exposed to seismic events (including beyond design basis seismic events) as part of the NRC's post-Fukushima actions resulting from Near-Term Task Force Recommendation 2.3: Seismic (Ref. A-17); and

- aging management programs to address material degradation due to aging in SSCs in the alternative pathway are addressed, for licensees that currently have extended operating licenses or will apply for such licenses in the future.

The application of these design and construction quality standards provides reasonable confidence regarding the robustness of the SSCs in the alternative drain pathway. Additionally, probabilistic information developed through earthquake operating experience and analysis of walkdown results confirms a low likelihood of failure at design basis seismic accelerations. Consideration of dynamic loading conditions or complete walkdowns of the credited pathway provides greater assurance of ruggedness for sites with higher seismic hazards.

Note: The revised method for seismic analysis of the alternative drain pathway presented below is intended to be used with the aerosol deposition models in Regulatory Position A-5.6.

<u>Revised Seismic Analysis of the Alternative Drain Pathway</u>

All licensees choosing this alternative should describe the code of record used for the main steamlines and the extent of quality assurance measures applied to the design, materials, and fabrication of the steamlines and attached piping. The description should include the alternate pathway identified to the main condenser and the basis for the reliability of this pathway. The licensee should also provide the following information for the site, as applicable for the site seismic hazard tier:

(1) If the piping and valves in the alternate pathway have been subjected to dynamic seismic analysis for the as-built configuration to a code of record (e.g., American Society of Mechanical Engineers (ASME) B31.1, "Power Piping" (Ref. A-18)), and the magnitude of the seismic response spectrum for the analysis equals or exceeds the licensee's safe-shutdown earthquake, then a description of the dynamic analysis provides sufficient justification.

(2) If the SSCs in the alternate pathway have not been subjected to dynamic seismic analysis to a code of record (e.g., ASME B31.1), and if the peak spectral acceleration of the ground motion response spectrum based on the licensee's most recent site-specific probabilistic seismic hazard is at most 0.4g, then the justification should include the following:

  - a discussion of seismic capacity and margin present in the relevant SSCs, including the condenser, based on their design code(s) of record,

  - insights from plant-specific seismic assessments performed as part of the Individual Plant Examination for External Events for relevant SSCs,

- a walkdown of a sample of the relevant SSCs, including the condenser, performed by knowledgeable licensee staff members, to verify that they have been constructed as designed, and

- confirmatory calculations for a sample of piping supports, to verify that they provide acceptable flexibility at terminal ends of piping and major branch connections.

The extent of the selected samples should be justified based on the plant-specific seismic hazard and quality assurance practices applied to design and fabrication. Details of the walkdown(s), including the qualifications of the licensee staff members performing them, should be retained in archival documentation.

(3) If the SSCs in the alternate pathway have not been subjected to dynamic seismic analysis to a code of record (e.g., ASME B31.1), and if the peak spectral acceleration of the ground motion response spectrum based on the licensee's most recent site-specific probabilistic seismic hazard is greater than 0.4g, then the justification should include the following:

- a discussion of seismic capacity and margin present in the relevant SSCs, including the condenser, based on their design code(s) of record,

- insights from the Individual Plant Examination for External Events as described above,

- walkdown(s) of the SSCs in the alternate pathway, including the condenser, performed by knowledgeable licensee staff members, to ensure that items adversely affecting the seismic capacity of relevant SSCs (e.g., loose or missing anchorages and degraded pipe supports) are identified and corrected, and

- confirmatory calculations for a sample of piping supports, to verify that they provide acceptable flexibility at terminal ends of piping and major branch connections.

The extent of the selected samples should be justified based on the plant-specific seismic hazard and quality assurance practices applied to design and fabrication. Details of the walkdown(s), including the qualifications of the licensee staff members performing them, should be retained in archival documentation.

A-5.6 For BWRs with Mark I, II, or III containment designs, aerosol deposition in horizontal volumes that meet Regulatory Position A-5.4 or A-5.5 may be credited as described below.[5] The NRC staff will consider aerosol deposition models for BWR designs other than those with Mark I, II, or III containment designs on a case-by-case basis.

A-5.6.1 <u>SAND2008-6601 model</u>: Section 6.4 of SAND2008-6601 describes an acceptable model for estimating the aerosol deposition between closed MSIVs and downstream of the MSIVs. Table A-1 provides the removal coefficients recommended by SAND2008-6601 and given in table 6-1 of that document.

5 The credit described in this regulatory position will supersede the aerosol settling estimates previously given in the NRC staff document AEB 98-03 (Ref. A-13) when Revision 1 of RG 1.183 is used.

Table A-1. BWR Main Steamline and Condenser Removal Coefficients

Time (hr)	Inboard (hr^{-1})	Between MSIVs (hr^{-1})	Outboard (hr^{-1})	Condenser (hr^{-1})
0–10	0.0	1.8	1.0	0.015
10+	0.0	1.0	0.7	0.012

A-5.6.2 Reevaluated AEB-98-03 with the multi-group method: Aerosol deposition removal coefficients for the main steamline piping between the MSIVs and downstream of the MSIVs may apply an updated AEB 98-03 use of the Stokes settling velocity physics parameters with the multi-group method. The method below computes both total effective aerosol removal efficiencies (TEAREs) (i.e., filter efficiencies) and equivalent removal coefficients λ (hr^{-1}).

When evaluating the Stokes settling velocity, use the aerodynamic mass median diameter (AMMD), d_a, based on a distribution directly measured from experiments to evaluate the settling velocity where the specific aerosol parameter distributions of shape factor, density, and volume-equivalent diameter do not need to be defined. Therefore, the Stokes settling velocity can be rewritten in terms of the aerodynamic diameter, d_a, as follows:

$$u_s = \frac{\rho_0 \cdot d_a^2 \cdot g \cdot C_s(d_a)}{18\mu} \qquad \text{(Equation A-1)}$$

where

ρ_0 = aerosol unit density = 1.0 g/cm^3,
d_a = aerosol aerodynamic diameter,
g = gravitational acceleration,
$C_s(d_a)$ = Cunningham slip factor as a function of d_a, and
μ = viscosity.

The document “State-of-the-Art Report on Nuclear Aerosols,” issued in 2009 (Ref. A-19), provides a summary of experimental observations from integral experiments involving irradiated fuel to infer characteristics of aerosols under light-water reactor severe accident conditions. The State-of-the-Art Report recommends the use of a log-normal distribution for aerosols in the RCS (AMMD 1.0 microns (µm) with a geometric standard deviation, σ_g, of 2.0), and provides PHÉBUS-Fission Product aerosol measurements in containment (AMMD of 3.0 µm and σ_g of 2.0). Considering the MHA LOCA modeling approach, which considers no pipe break and where the deposition properties after reflood are based on the characteristics of the RCS and containment aerosol (i.e., the approach considers the effects of an active emergency core cooling system), the methods in Regulatory Positions A-5.6.2 and A-5.6.3 should assume a log-normal aerosol diameter distribution with an AMMD of 2.0 µm and σ_g of 2.0. Assume as fixed values a $C_s(d_a)$ of 1 and a viscosity of 1.93x10^{-5} Pascal-second. At least 10,000 trials are necessary to develop a settling velocity distribution dataset. Note that while the NRC memorandum of Reference A-20 addresses the methods discussed in Regulatory Positions A-5.6.2 and A-5.6.3, it does not establish regulatory positions. For example, regarding Reference A-20, this memorandum does not endorse input parameters such as the AMMD, assumed in example calculations, and statements on the validity of the existing 20-group method.

The multi-group method should include the following assumptions and steps to estimate removal coefficients:

a. Discretize the settling velocity dataset into at least 2,000 equal-width groups. Assign a relative probability to each group by dividing the number of data points within each group by the sample size (e.g., 10,000 trials) to determine the group probabilities. Identify the midpoint of each group to represent the settling velocity for that group.

b. Compute each group's aerosol filter efficiency using the following method. By rearranging equations 2, 3 and 4 from Reference A-13, the filter efficiency, η_{filt}, is computed by using the group settling velocity, the settling area, the volumetric flow rate, and the volume of the well-mixed region being modeled as follows:

$$\eta_{filt} = 1 - \frac{C_{out}}{C_{in}} = 1 - \frac{1}{1+\frac{\lambda * V}{Q}} = 1 - \frac{1}{1+\frac{u_s * A}{Q}} \qquad \text{(Equation A-2)}$$

where

η_{filt} = removal, or filter, efficiency,
u_s = settling velocity (ft/hr),
A = settling area (ft^2),
C_{out} = outgoing concentration of nuclides in the pipe segment volume,
C_{in} = initial concentration of nuclides in the pipe segment volume,
Q = volumetric flow rate into pipe segment volume (ft^3/hr), and
λ = equivalent removal coefficient (hr^{-1}).

Account for the effect of the changing settling velocity distribution in the downstream volumes by adjusting the downstream volume efficiencies by multiplying them by the prior volume aerosol filter removal efficiency.

c. Compute the TEAREs (i.e., filter efficiencies) and equivalent removal coefficients, λ (hr^{-1}), for a credited volume by the following method. Compute the probability-weighted aerosol filter efficiency by multiplying the aerosol filter efficiency by the group probability from step 1. Then sum all the probability-weighted aerosol removal efficiencies to obtain the TEARE. By solving for λ in equation A-2 the removal coefficients are computed to yield:

$$\lambda = \frac{-\eta_{filt} * Q}{(\eta_{filt} - 1) * V} \qquad \text{(Equation A-3)}$$

where

η_{filt} = TEAREs,
Q = volumetric flow rate into credited volume, and
V = well-mixed pipe free volume.

A-5.6.3 Numerical Integration: Aerosol deposition removal coefficients for the main steamline piping between the MSIVs and downstream of the MSIVs may apply the method's use of the Stokes settling velocity and physics parameters of Regulatory Position A-5.6.2 with the numerical integration method. The equation for a normalized number distribution ($n(d_a)$) of particles of aerodynamic diameter (d_a) is given (Ref. A-21) as follows:

$$n(d_a) = \frac{1}{d_a\sqrt{2\pi}\ln(\sigma_g)} Exp\left[-\frac{\ln\left(\frac{d_a}{d_g}\right)^2}{2\ln(\sigma_g)^2}\right] \quad \text{(Equation A-4)}$$

where

σ_g= geometric standard deviation, and
d_g= geometric mean (which, for a log-normal distribution, is the same as the median diameter).

According to the Hatch-Choate equations, the AMMD is related to the median diameter (d_g), in meters, as follows:

$$d_g = AMMD\ Exp[-3\ln(\sigma_g)^2] \quad \text{(Equation A-5)}$$

Discretize the range of particle diameters, d_a, from $1\text{x}10^{-9}$ μm to $1\text{x}10^{-3}$ μm into 150 groups. For each group, apply equation A-4 to compute the normalized number distribution, ($n(d_a)$). Then, for each discretized group, (1) compute its settling velocity by applying equation A-1, (2) use equation 3 from Reference A-20 to compute inboard and outboard concentrations of particles leaving the volumes, (3) sum up the inboard and outboard concentrations using an appropriate numerical integration technique (such as the trapezoidal method), and (4) use equation A-2 to then compute the filter efficiencies. Finally, use equation A-3 to convert filter efficiencies, η_{filt}, into removal coefficients (in units of hr^{-1}).

A-5.6.4 Aerosol deposition removal coefficients for the condenser using a multi-group method and numerical integration are acceptable and will be evaluated on a case-by-case basis.

A-5.7 Reduction of the amount of released elemental iodine by plateout deposition on steam system piping may be credited, but the amount of reduction in concentration allowed will be evaluated on a case-by-case basis. The model should assume well-mixed volumes. Reference A-22 provides guidance on an acceptable model.

A-5.8 Reduction of the amount of released organic iodine (e.g., using the Brockman-Bixler model in RADTRAD (Ref. A-5)) should not be credited.

A-5.9 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in Regulatory Position A-5.4, then the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground-level release.

A-5.10 Hold-up and dilution of MSIV leakage releases into the turbine building should not be assumed.

A-6. Containment Purging

The licensee should analyze the radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure. If the installed containment purging capabilities are maintained for the purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with the consequences postulated for other fission product release paths to determine the total calculated radiological consequences of the LOCA. The licensee may take into account the reduction in the amount of radioactive material released through ESF filter systems using the guidance in RG 1.52 (Ref. A-6).

APPENDIX A

REFERENCES[1]

A-1. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

A-2. U.S. Nuclear Regulatory Commission (NRC), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Washington, DC.

A-3. NRC, NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," Washington, DC, July 1996 (ML100130305).

A-4. NRC, NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," Washington, DC, June 1993 (ML063480542).

A-5. NRC, NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," Washington, DC, April 1998 (ML15092A284); NUREG/CR-7220, "SNAP/RADTRAD 4.0: Description of Models and Methods," Washington, DC, June 2016 (ML16160A019).

A-6. NRC, Regulatory Guide 1.52, Revision 4, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Washington, DC, September 2012 (ML12159A013).

A-7. NRC, NUREG-1966, "Final Safety Evaluation Report Related to the Certification of the Economic Simplified Boiling-Water Reactor Standard Design," Washington, DC, April 2014 (ML14100A304).

A-8. NRC, Information Notice 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Washington, DC, September 19, 1991 (ML031190264).

A-9. NRC, Information Notice 88-76, "Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control," Washington, DC, September 19, 1988 (ML031150101).

A-10. NRC, NUREG-0737, "Clarification of TMI Action Plan Requirements," Washington, DC, November 1980 (ML102560051).

A-11. NRC, NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," Washington, DC, February 1995 (ML041040063).

1 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

A-12. Sandia National Laboratories, SAND2008-6601, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," Albuquerque, New Mexico, October 2008 (ML083180196; ML113400138 (errata)).

A-13. NRC, Accident Evaluation Branch (AEB)-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term," Washington, DC, December 9, 1998 (ML011230531).

A-14. NRC, Regulatory Guide 1.29, "Seismic Design Classification for Nuclear Power Plants," Washington, DC.

A-15. NRC, letter to T.A. Green, Boiling Water Reactor Owners' Group (BWROG) Projects, "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems, September 1993,'" Washington, DC, March 3, 1999 (ML010640286).

A-16. BWROG, NEDC-31858P, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Birmingham, Alabama, September 1993 (Not publicly available).

A-17. Leeds, E.J. and Johnson, M. R., NRC, letter to all power reactor licensees and holders of construction permits in active or deferred status, "Request for Information Pursuant to Title 10 of the *Code of Federal Regulations* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," Washington, DC, March 12, 2012 (ML12053A340).

A-18. American Society of Mechanical Engineers (ASME), ASME B31.1, "Power Piping," New York, NY, 2001.

A-19. Allelein, H.-J.A., NEA/CSNI/R(2009)5, "State-of-the-Art Report on Nuclear Aerosols," Nuclear Energy Agency, Committee on the Safety of Nuclear Installations, Paris, 2009.

A-20. Dickson, E., NRC, internal memorandum to K. Hsueh, "Technical Basis for Draft RG 1.183 Revision 1 (2021) Re-evaluated AEB-98-03 Settling Velocity Method, the Multi-Group Method, and the Numerical Integration Method," Washington, DC, July 29, 2021 (ML21141A006).

A-21. Williams, M.M.R., *Aerosol Science Theory and Practice*, Pergamon Press, New York, New York, 1991.

A-22. J.E. Cline and Associates, Inc., "MSIV Leakage Iodine Transport Analysis," letter report, Rockville, Maryland, March 26, 1991 (ML003683718).

APPENDIX B

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a fuel handling accident at a light-water reactor. These assumptions supplement the guidance in the main body of this guide.

B-1. Source Term

Regulatory Position 3 of this guide provides acceptable assumptions regarding core inventory and the release of radionuclides from the fuel. The following assumptions also apply:

B-1.1 The number of fuel rods assumed to be damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples); the height of the drop; and the compression, torsion, and shear stresses on the irradiated fuel rods. The analysis should also consider damage to adjacent fuel assemblies, if applicable (e.g., for events over the reactor vessel).

B-1.2 The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.

B-1.3 The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This regulatory position and those in B-2 use, in part, the transport models discussed in References B-1 and B-2.

All the gap activity in the damaged rods is assumed to be released over two phases:

- Phase 1—the instantaneous release from the rising bubbles (from start of accident to 2 hours). Elemental iodine and organic iodine are conservatively assumed to be in vapor form. Elemental iodine is subsequently decontaminated by passage through the overlying pool of water into the building atmosphere.

- Phase 2—the protracted release due to re-evolution as elemental iodine (starts at 2 hours and ends at 30 days). CsI is conservatively assumed to completely dissociate into the pool water. Because of the low pH of the pool water, CsI (as well as Phase 1 absorbed elemental iodine within the pool) slowly re-evolves as elemental iodine into the building atmosphere.

B-1.4 The radioactive material available for release is assumed to be from the assemblies with the peak inventory. The fission product inventory for the peak assembly represents an upper limit value. The inventory should be calculated assuming the maximum achievable operational power history and burnup. These parameters should be examined to maximize fission product inventory. This inventory calculation should include appropriate assembly peaking factors.

B-2. Phase 1 Release—Initial Gaseous Release and Water Depth

The elemental iodine decontamination factor (DF) is a function of bubble size and rise time through the water column, both of which are functions of fuel pin pressure. If the water depth is between 19 and 23 feet, the DF for elemental iodine can be computed based on a best-estimate rod pin pressure for the limiting fuel rods in the reactor core at the most limiting time in life. The time period between reactor shutdown and the movement of fuel may be used to compute radioactive decay and reduced decay power. The internal gas temperature, and thus the pin pressure, may be determined using the limiting pool water temperature near the fuel rods, and basing these values on a full-core offload.

For water depths between 19 and 23 feet, the elemental iodine DF based on pin pressure is computed using the following equations:

$$DF_I = 81.046e^{0.305\left(t/d\right)}, \qquad \text{(Equation B-1)}$$

$$t = 9.2261e^{-6\text{x}10^{-4}*x}, \qquad \text{(Equation B-2)}$$

$$d = -0.0002 * x + 1.0009, \qquad \text{(Equation B-3)}$$

where *t* is the bubble rise time in seconds, computed as a function of pin pressure, *x*, in pounds per square inch gauge (psig), and *d* is the bubble diameter in centimeters, computed as a function of pin pressure, *x* (psig).

If the depth of water is not between 19 and 23 feet, the DF will need to be determined on a case-by-case basis. The DF for organic iodine is assumed to be 1.

B-3. Phase 2 Release—Re-evolution Release

The re-evolution calculation results in a simple exact transient solution. It has the flexibility to account for the effect of potential filtration and other removal mechanisms. The following site-specific and general parameters are needed:

a. V_{pool} = total pool free volume,
b. S_{pool} = total pool surface area,
c. Q_{recirc} = volumetric flow of recirculation system (to evaluate effects of filtration),
d. $N_{I\text{-}131gap}$ = fuel pin radioactive iodine in gap (moles),
e. $N_{I\text{-}127gap}$ = fuel pin nonradioactive iodine in gap (moles),
f. K_L = mass transfer coefficient, $3.66\text{x}10^{-6}$ m/s (Ref. B-1), and
g. pH = bounding design acidity value of the pool.

Note: For this approach, V_{pool}, S_{pool}, K_L, and Q_{recirc} must use consistent units. (To calculate concentrations in moles per liter (M), V_{pool} must be converted to liters.)

<u>Calculation Sequence:</u>

1. Calculate amount of iodine (radioactive and nonradioactive) in the fuel pin gap using tables 3 and 4 from the main body of this guide.

2. Calculate volatile iodine fraction in pool.

3. Calculate removal coefficients.

4. Evaluate release as either (a) an overall release (neglecting time), or (b) a time-dependent release.

<u>Step 1—Calculate amount of iodine in the fuel pin gap using tables 3 and 4 in the main body of this guide.</u>

Both the radioactive and the nonradioactive iodine (e.g., I-131 and I-127) in the pool affect the radioactive iodine evolution. The calculations operate on moles, so iodine isotope quantities must be converted to moles.

In the example equations that follow, I-131 is used to represent the radioactive iodine and I-total is used to represent the stable iodine; however, all radioactive and nonradioactive iodine should be considered. For a given mass of iodine, the number of moles of iodine can be calculated from the mass, *m*, in grams (g) and its atomic weight, *M*, as follows:

$$N_{\text{I-131}} = \left(\frac{m_{\text{I-131}}(\text{g})}{M_{\text{I-131}}\left(\text{g}/_{\text{mol}}\right)} \right), \qquad \text{(Equation B-4)}$$

$$N_{\text{I-total}} = \left(\frac{m_{\text{I-}total}(\text{g})}{M_{\text{I-total}}\left(\text{g}/_{\text{mol}}\right)} \right). \qquad \text{(Equation B-5)}$$

Alternatively, for radioactive materials, the number of moles can be calculated from the activity in becquerels (Bq):

$$N_{\text{I-131}} = \left(\frac{A_{\text{I-131}}\left(\text{dis}/_{\text{s}}\right)}{\lambda_{\text{I-131}}\left(\text{dis}/_{\text{atom·s}}\right)} \right). \qquad \text{(Equation B-6)}$$

Activities in curies (Ci) must be converted to becquerels (1 Ci = 3.7×10^{10} Bq).

The radioactive iodine concentration can be found using radiological decay formulas that account for time before fuel movement. If this is done, the activity of the other iodine isotopes at the time before fuel movement should be added to the I-131 activity.

<u>Step 2—Calculate volatile iodine fraction in pool.</u>

The next step is to determine the fraction of iodine atoms in the pool that are in I_2 (volatile) form:

- Calculate the total concentrations in the pool:

  C_t = total I concentration (M) (moles I atoms / L) = ($N_{\text{I-total\_gap}} + N_{\text{I-131gap}}$) / V_{pool}.
  (Equation B-7)

  Note: V_{pool} must be converted to liters for concentrations to be calculated in moles/liter.

- Calculate the H^+ concentration:

  $$C_h = [H^+] = 10^{-pH}. \qquad \text{(Equation B-8)}$$

- Calculate the $[I_2]/[I-]^2$ concentration ratio, R_i (Ref. B-2):[1]

$$R_i = [I_2]/[I-]^2 = C_h^2 / (6.05\times10^{-14} + 1.47\times10^{-9}\, C_h). \qquad \text{(Equation B-9)}$$

- Calculate the fraction of I atoms in I_2 form:

  - First evaluate B_m (negative B for the quadratic equation below):

$$B_m = 4\, C_t + 1 / R_i. \qquad \text{(Equation B-10)}$$

  - Then evaluate the volatile fraction, X_e (fraction of I atoms in I_2 form):

$$X_e = (\, B_m - \sqrt{B_m^2 - 16\, C_t^2}\,) / (\, 4\, C_t). \qquad \text{(Equation B-11)}$$

<u>Step 3—Calculate applicable removal coefficients.</u>

The evolution removal coefficient, λ_e, is calculated using the mass transfer coefficient, the pool surface-to-volume ratio, and the fraction of I that is in I_2 form:

$$\lambda_e = K_L\, X_e\, S_{pool} / V_{pool} \qquad \text{(Equation B-12)}$$

The removal rate is reduced to account for the fraction of iodine that is volatile and thus available to evolve to the gas space. This evolution rate applies to both nonradioactive and radioactive iodine.

<u>Step 4—Evaluate release as an overall release.</u>

The removal coefficient is used to model the time-dependent concentration of radionuclides released from the pool as follows:

$$Q_e = \lambda_e\, V_{pool} \qquad \text{(Equation B-13)}$$

a. If recirculation filtration is credited, λ_f is used.
b. Alternatively, one can model a loop and filter instead of using λ_f.

B-4. Noble Gases and Particulates

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., the DF is 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., the DF is infinite).

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1 The combined speciation rate is from NUREG/CR-5950, “Iodine Evolution and pH Control,” issued December 1992 (Ref. B-3).

B-5. Fuel Handling Accidents within the Fuel Building

For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff:

B-5.1 The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period for the initial fuel gap gas release, which accounts for time-independent releases from the fuel pool. The release rate is generally assumed to be a linear or exponential function over this time period. Time-dependent releases from the fuel pool due to the re-evolution of iodine are to be considered releases directly from the pool to the environment outside the fuel building.

B-5.2 Engineered-safety-feature (ESF) filtration systems may reduce the amount of radioactive material released from the fuel pool. Such a reduction may be taken into account using the guidance in Regulatory Guide (RG) 1.52, Revision 4, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," issued September 2012 (Ref. B-4). The radioactivity release analyses should determine and account for delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system.[2]

B-5.3 The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede streamflow between the surface of the pool and the exhaust plenums.

B-6. Fuel Handling Accidents within the Containment

For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff:

B-6.1 If the containment is isolated[3] during fuel handling operations, no radiological consequences need to be analyzed.

B-6.2 If the containment is open during fuel handling operations but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed for the isolated pathway.

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2 These analyses should consider the time needed for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.

3 Containment isolation does not imply containment integrity as defined by technical specifications for non-shutdown modes. The term "isolation" is used here collectively to encompass both containment integrity and containment closure, which is typically in place during shutdown periods. For isolation to be credited in the analysis, the technical specifications should address the appropriate form of isolation.

B-6.3 If the containment is open during fuel handling operations (e.g., a personnel air lock or equipment hatch is open),[4] the radioactive material that escapes from the reactor cavity pool to the containment is assumed to be released to the environment over a 2-hour period for the initial fuel gap gas release, which accounts for time-independent releases from the reactor cavity. The release rate is generally assumed to be a linear or exponential function over this period. Time-dependent releases from the reactor cavity pool due to the re-evolution of iodine are to be considered releases directly from the pool to the environment outside the containment.

B-6.4 A reduction in the amount of radioactive material released from the containment by ESF filtration systems may be taken into account using the guidance of RG 1.52 (Ref. B-4). The radioactivity release analyses should determine and account for delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system.[5]

B-6.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50 percent of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede streamflow between the surface of the reactor cavity and the exhaust plenums.

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4 Technical specifications that allow such operations usually include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such administrative controls generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses generally should not credit this manual isolation.

5 These analyses should consider the time needed for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.

APPENDIX B

REFERENCES[1]

B-1. U.S. Nuclear Regulatory Commission (NRC), “Re-evaluation of the Fission Product Release and Transport for the Design-Basis Accident Fuel Handling Accident” (ML19248C647), Enclosure 4 to internal memorandum from M.J. Case to M.X. Franovich, “Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-Basis Accident Fuel Handling Accident,” Washington, DC, November 23, 2019 (ML19114A117 (package)).

B-2. Dickson, E., NRC, internal memorandum to K. Hsueh, “Example Calculation of Re-evaluated Fuel Handling Accident Fission Product Transport Model for Draft RG 1.183 Revision 1 (2021),” Washington, DC, August 30, 2021 (ML21190A040).

B-3. NRC, NUREG/CR-5950, “Iodine Evolution and pH Control,” Washington, DC, December 1992 (ML063460464).

B-4. NRC, Regulatory Guide 1.52, Revision 4, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Washington, DC, September 2012 (ML12159A013).

1 Publicly available NRC published documents are available electronically through the NRC Library on the NRC’s public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC’s Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

APPENDIX C

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BOILING-WATER REACTOR ROD DROP ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a rod drop accident at a boiling-water reactor. These assumptions supplement the guidance in the main body of this guide.

C-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory. The fission product release from the breached fuel to the coolant is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. In addition to the combined fission product inventory (steady-state gap plus transient release), the release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting, and on the assumption that 100 percent of the noble gases and 50 percent of the iodines contained in that fraction are released to the reactor coolant.[1]

C-2. If no or minimal fuel breach[2] is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by the technical specifications (typically a pre-accident spike of 4.0 microcuries per gram dose equivalent (DE) iodine (I)-131).

C-3. The assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material from the fuel and the reactor coolant are as follows:

C-3.1 The activity released from the fuel from the gap and/or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

C-3.2 Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.

C-3.3 Of the activity released from the reactor coolant within the pressure vessel, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the remaining radionuclides are assumed to reach the turbine and condensers.

C-3.4 Of the activity that reaches the turbine and condenser, 100 percent of the noble gases, 10 percent of the iodine, and 1 percent of the particulate radionuclides are assumed available for release to the environment. The turbine and condensers leak to the environment as a ground-level release at a rate of 1 percent per day[3] for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or hold-up within the turbine building. Radioactive decay during hold-up in the turbine and condenser may be assumed.

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1 Calculated values of the combined release (gap activity plus fuel melt) are limited to a total of 1.0.

2 "Minimal fuel breach" is defined for use in this appendix as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel breach or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the DE I-131, only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

3 If there are forced flowpaths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leak rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by offgas or standby gas treatment, will be considered on a case-by-case basis.

C-3.5 In lieu of the transport assumptions in Regulatory Positions C-3.2 through C-3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers, based on a review of the minimum transport time from the pressure vessel to the first main steam isolation valve and the closure time for this valve.

C-3.6 The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95 percent cesium iodide as an aerosol, 4.85 percent elemental iodine, and 0.15 percent organic iodide. The release from the turbine and condenser should be assumed to be 97 percent elemental and 3 percent organic.

APPENDIX D

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BOILING-WATER REACTOR MAIN STEAMLINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a main steamline break accident at a boiling-water reactor. These assumptions supplement the guidance in the main body of this guide.

Source Term

D-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel.

D-2. If no or minimal fuel breach[1] is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specifications (TS). The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the standard TS for the nuclear steam supply system vendor:

D-2.1 The concentration that is the maximum value permitted (typically 4.0 microcuries per gram (μCi/g) dose equivalent (DE) iodine (I)-131) and corresponds to the conditions of an assumed pre-accident spike.

D-2.2 The concentration that is the maximum equilibrium value permitted for continued full-power operation (typically 0.2 μCi/g DE I-131).

D-3. The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. Noble gases should be assumed to enter the steam phase instantaneously.

Transport

D-4. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

D-4.1 The main steamline isolation valves should be assumed to close in the maximum time allowed by TS.

D-4.2 The total mass of coolant released should be assumed to be the amount in the steamline and connecting lines at the time of the break, plus the amount that passes through the valves before closure.

D-4.3 All radioactivity in the released coolant should be assumed to be released to the environment instantaneously as a ground-level release. No credit should be assumed for plateout, hold-up, or dilution within facility buildings.

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1 "Minimal fuel breach" is defined for use in this appendix as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining DE I-131, only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

D-4.4 The iodine species released from the main steamline should be assumed to be 95 percent cesium iodide as an aerosol, 4.85 percent elemental iodine, and 0.15 percent organic iodide.

APPENDIX E

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR STEAM GENERATOR TUBE RUPTURE ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a steam generator tube rupture (SGTR) accident at a pressurized-water reactor. These assumptions supplement the guidance in the main body of this guide.

Source Term

E-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel.

E-2. If no or minimal fuel breach[1] is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specifications (TS). Two cases of iodine spiking should be assumed:

E-2.1 A reactor transient has occurred before the postulated SGTR and has raised the primary coolant iodine concentration to the maximum value permitted at full-power operations by the TS (typically 60 microcuries per gram (μCi/g) dose equivalent (DE) iodine (I)-131). This is the pre-accident iodine spike case.

E-2.2 The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in the TS (typically 1.0 μCi/g DE I-131). This is the concurrent iodine spike case. A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel pins assumed to have defects.

E-3. The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

E-4. The specific activity in the steam generator liquid at the onset of the SGTR is at the maximum value permitted by secondary activity TS (typically 0.1 μCi/g).

E-5. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental iodine and 3 percent organic iodide.

Transport

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1 "Minimal fuel breach" is defined for use in this appendix as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining DE I-131, only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

E-6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

E-6.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the TS. The primary-to-secondary leak rate at later stages of the transient may be reduced if justified by plant-specific design and engineering analyses. The leakage should be apportioned between affected and unaffected steam generators in a manner that maximizes the calculated dose.

E-6.2 The density used in converting volumetric leak rates (e.g., in gallons per minute) to mass leak rates (e.g., in pounds mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rates in the TS. These tests are typically based on cool liquids. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

E-6.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature in the bulk of the primary system is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated. The release of radioactivity from the affected steam generator should be assumed to continue until shutdown cooling is in operation and releases from the steam generator have been terminated, or the steam generator is isolated from the environment so that no release is possible, whichever occurs first.

E-6.4 All noble gas radionuclides released from the primary system should be assumed to be released to the environment without reduction or mitigation.

E-6.5 The transport model described in this section should be used for iodine and particulate releases from the steam generators. Figure E-1 illustrates this model, which is summarized as follows:

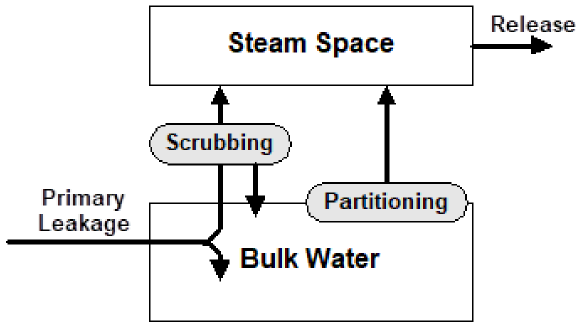

Figure E-1. Transport model

E-6.5.1 A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant (bulk water in figure E-1).

For the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage, discussed in Regulatory Position E-6.1, can be assumed to mix with the secondary water without flashing during periods of total tube submergence.

E-6.5.2 The leakage in the affected steam generator that immediately flashes to vapor will rise through the secondary water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," issued January 1978 (Ref. E-1), during periods of total submergence of the tubes.

E-6.5.3 The leakage in the affected steam generator that does not immediately flash is assumed to mix with the secondary water.

E-6.5.4 The radioactivity in the secondary water is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient.[2] A partition coefficient of 100 may be assumed for iodine. The retention of noniodine particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

E-6.6 During periods of steam generator dryout, all primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation.

Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-2). If the tubes are uncovered, a portion of the primary-to-secondary leakage will flash and atomize, based on the thermodynamic conditions in the reactor and secondary coolant, and will be released to the environment with no mitigation. The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction) needs to be considered. The impact of restoration strategies described in emergency operating procedures for steam generator water levels should be evaluated.

2 In this appendix, the partition coefficient is defined as follows:

$$PC = \frac{mass\ \ of\, I_2\ per\ unit\ mass\ of\ liquid}{mass\ \ of\, I_2\ per\ unit\ mass\ of\ gas}$$

APPENDIX E

REFERENCES[1]

E-1. U.S. Nuclear Regulatory Commission (NRC), NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," Washington, DC, January 1978 (ML19269F014).

E-2. NRC, Information Notice 88-31, "Steam Generator Tube Rupture Analysis Deficiency," Washington, DC, May 25, 1988 (ML031150151).

1 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

APPENDIX F

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR MAIN STEAMLINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a main steamline break (MSLB) accident at a pressurized-water reactor. These assumptions supplement the guidance in the main body of this guide.

Source Term

F-1. Regulatory Position 3 of this regulatory guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel.

F-2. If no or minimal fuel breach[1] is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications (TS). Two cases of iodine spiking should be assumed:

F-2.1 A reactor transient has occurred before the postulated MSLB and has raised the primary coolant iodine concentration to the maximum value permitted by the TS (typically 60 microcuries per gram (μCi/g) dose equivalent (DE) iodine (I)-131). This is the pre-accident iodine spike case.

F-2.2 The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value specified in the TS (typically 1.0 μCi/g DE I-131). This is the concurrent iodine spike case. A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap assumed to have defects.

F-3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest-worth control rod is stuck at its fully withdrawn position.

F-4. The specific activity in the steam generator liquid at the onset of the MSLB should be assumed to be at the maximum value permitted by secondary activity TS (typically 0.1 μCi/g DE I-131).

F-5. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental iodine and 3 percent organic iodide. These fractions apply both to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

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1 "Minimal fuel breach" is defined for use in this appendix as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining DE I-131, only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

Transport

F-6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

F-6.1 The secondary water in the faulted[2] steam generator is assumed to rapidly blow down to the environment. The duration of the blowdown is obtained from thermal-hydraulic analysis codes. The activity in the faulted steam generator secondary water is assumed to be released to the environment without mitigation.

F-6.2 For facilities that have not implemented alternative repair criteria, the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the TS. For facilities with traditional steam generator specifications (both per generator and for the total of all generators), the leakage should be apportioned between faulted and unaffected steam generators in a manner that maximizes the calculated dose. For example, for a four-loop facility with a limiting condition for operation of $1.9x10^3$ liters per day (500 gallons per day) for any one generator, not to exceed 3.8 liters per minute (1 gallon per minute) from all generators, it would be appropriate to assign $1.9x10^3$ liters per day (500 gallons per day) to the faulted generator and $1.2x10^3$ liters per day (313 gallons per day) to each of the unaffected generators.

For facilities that have implemented alternative repair criteria, the primary-to-secondary leak rate in the faulted steam generator should be assumed to be the maximum accident-induced leakage derived from the repair criteria and burst correlations. For the unaffected steam generators, the leak rate limiting condition for operation specified in the TS is equally apportioned between the unaffected steam generators.

F-6.3 The density used in converting volumetric leak rates (e.g., in gallons per minute) to mass leak rates (e.g., in pounds mass per hour) should be consistent with the basis of the parameter being converted. The leak rate correlations for alternative repair criteria are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate TS are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

F-6.4 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature in the bulk of the primary system is less than 100 degrees Celsius (212 degrees Fahrenheit). The primary-to-secondary leak rate at later stages of the transient may be reduced if justified by plant-specific design and engineering analyses. The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

F-6.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

F-6.6 The transport model described in this section should be used for iodine and particulate releases from the steam generators.

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2 In this appendix, “faulted” refers to the state of the steam generator in which the secondary side has been depressurized by an MSLB, in such a way that protective system response (main steamline isolation, reactor trip, safety injection, etc.) has occurred.

F-6.6.1 For the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.

F-6.6.2 The radioactivity in the secondary water of the unaffected generators is assumed to become vapor at a rate that is a function of the steaming rate and the partition coefficient. A partition coefficient of 100 may be assumed for iodine. The retention of particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

F-6.6.3 The primary-to-secondary leakage to the faulted steam generator is assumed to flash to vapor and be released to the environment with no mitigation.

Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. F-1). If the tubes are uncovered, a portion of the primary-to-secondary leakage will flash and atomize, based on the thermodynamic conditions in the reactor and secondary coolant, and will be released to the environment with no mitigation. The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction) needs to be considered. The impact of restoration strategies described in emergency operating procedures for steam generator water levels should be evaluated.

APPENDIX F

REFERENCES[1]

F-1 U.S. Nuclear Regulatory Commission, Information Notice 88-31, “Steam Generator Tube Rupture Analysis Deficiency,” Washington, DC, May 25, 1988 (ML031150151).

1 Publicly available NRC published documents are available electronically through the NRC Library on the NRC’s public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC’s Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

APPENDIX G

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR LOCKED ROTOR ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a locked rotor accident at a pressurized-water reactor. These assumptions supplement the guidance in the main body of this guide.

Source Term

G-1. Regulatory Position 3 of this regulatory guide provides assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel.

G-2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required, as the consequences of this event are bounded by the consequences projected for the main steamline break outside containment.

G-3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

G-4. The chemical form of radioiodine released from the fuel should be assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental iodine and 3 percent organic iodide. These fractions apply both to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

Transport

G-5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

G-5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications (TS). The primary-to-secondary leak rate at later stages of the transient may be reduced if justified by plant-specific design and engineering analyses. The leakage should be apportioned between the steam generators in a manner that maximizes the calculated dose.

G-5.2 The density used in converting volumetric leak rates (e.g., in gallons per minute) to mass leak rates (e.g., in pounds mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate TS. These tests are typically based on cool liquids. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

G-5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature in the bulk of the primary system of the leakage is less than 100 degrees Celsius (212 degrees Fahrenheit). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

G-5.4 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

G-5.5 The transport model described in Regulatory Positions E-6.5 and E-6.6 of appendix E to this guide should be used for iodine and particulates.

APPENDIX H

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PRESSURIZED-WATER REACTOR CONTROL ROD EJECTION ACCIDENT

This appendix provides assumptions acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for evaluating the radiological consequences of a control rod ejection accident at a pressurized-water reactor. These assumptions supplement the guidance in the main body of this guide. Two release paths are considered: (1) release via containment leakage and (2) release via the secondary plant. Each release path is evaluated independently, as if it were the only pathway available. The consequences of this event are acceptable if the dose from each path considered separately is less than the acceptance criterion in table 7 in this guide.

Source Term

H-1. Regulatory Position 3 of this guide provides assumptions acceptable to the NRC staff regarding core inventory. The fission product release from the breached fuel to the coolant is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. In addition to the combined fission product inventory (steady-state gap plus transient release), the release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting, and on the assumption that 100 percent of the noble gases and 50 percent of the iodines contained in that fraction are released to the reactor coolant.[1]

H-2. If no fuel breach is postulated for the limiting event, a radiological analysis is not required, as the consequences of this event are bounded by the consequences projected for the maximum hypothetical loss-of-coolant accident, main steamline break, and steam generator tube rupture.

H-3. In the case of the first release path, 100 percent of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the case of the second release path, 100 percent of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

H-4. The chemical form of radioiodine released to the containment atmosphere should be assumed to be 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. If containment sprays do not actuate or are terminated before sump water is accumulated, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids created during the control rod ejection accident event (e.g., pyrolysis and radiolysis products). With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

H-5. Iodine releases from the steam generators to the environment should be assumed to be 97 percent elemental iodine and 3 percent organic iodide.

1 Calculated values of the combined release (gap activity plus fuel melt) are limited to a total of 1.0.

Transport from Containment

H-6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows:

H-6.1 A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.

H-6.2 The containment should be assumed to leak at the leak rate incorporated in the technical specifications (TS) at peak accident pressure for the first 24 hours and at 50 percent of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the TS for containment leak testing. Leakage from sub-atmospheric containments is assumed to be terminated when the containment is brought to a sub-atmospheric condition, as defined in TS.

Transport from Secondary System

H-7. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the secondary system are as follows:

H-7.1 A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the TS should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated. The primary-to-secondary leak rate at later stages of the transient may be reduced if justified by plant-specific design and engineering analyses.

H-7.2 The density used in converting volumetric leak rates (e.g., in gallons per minute) to mass leak rates (e.g., in pounds mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate TS. These tests are typically based on cooled liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gram per cubic centimeter (62.4 pounds mass per cubic foot).

H-7.3 All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.

H-7.4 The transport model described in Regulatory Positions E-6.5 and E-6.6 of appendix E to this guide should be used for iodine and particulates.

APPENDIX I

ANALYTICAL TECHNIQUE FOR CALCULATING FUEL-DESIGN OR PLANT-SPECIFIC STEADY-STATE FISSION PRODUCT RELEASE FRACTIONS FOR NON-LOSS-OF-COOLANT ACCIDENT EVENTS

This appendix provides an acceptable analytical technique for calculating steady-state fission product release fractions residing in the fuel rod void volume (plenum and pellet-to-cladding gap), based on either specific fuel rod designs or more realistic fuel rod power histories. This analytical procedure was used, along with bounding fuel rod power histories, to calculate the release fractions listed in tables 3 and 4 of Regulatory Position 3.2 in the main body of this guide. Lower release fractions are achievable using less aggressive rod power histories or less limiting fuel rod designs (e.g., 17x17 versus 14x14 fuel rod designs). The analytical technique outlined in this section is one acceptable means of calculating maximum steady-state release fractions.

Steady-state gap inventories represent radioactive fission products generated during normal steady-state operation that have diffused within the fuel pellet, have been released into the fuel rod void space (i.e., rod plenum and pellet-to-cladding gap), and are available for release upon fuel rod cladding failure. Given the continued accumulation of long-lived radioactive isotopes and the inevitable decay of short-lived radioactive isotopes, the most limiting time in life (i.e., maximum gap fraction) for a particular radioactive isotope varies with fuel rod exposure and power history. The analytical technique described in this appendix specifies the use of fuel rod power profiles based on core operating limits or limiting fuel rod power histories. In addition, this analytical technique produces a composite worst time-in-life (i.e., maximum gap fraction for each radioactive isotope). Therefore, the steady-state fission product gap inventories calculated using this analytical approach will be significantly larger than realistic fuel rod or core-average source terms. One means of capturing more realism in the calculation of the steady-state release fractions would be to calculate burnup-dependent release fractions for each radionuclide. The use of such means will be considered on a case-by-case basis.

The U.S. Nuclear Regulatory Commission (NRC) codeveloped the Fuel Analysis under Steady-State and Transients (FAST) (formerly FRAPCON and FRAPTRAN) fuel rod thermal-mechanical fuel performance code to perform independent audit calculations for licensing activities. While calibrated and validated against a large empirical database, FAST and its predecessors are not NRC-preapproved codes and may not be used to calculate plant-specific, fuel-specific, or cycle-specific gap inventories that are in accordance with the acceptable analytical procedure below without further justification.

The analytical technique used to calculate steady-state gap inventories should have the following attributes:

I-1. For stable, long-lived radioactive isotopes, such as krypton (Kr)-85, an NRC-approved fuel rod thermal-mechanical performance code with established modeling uncertainties should be used to predict the integral fission gas release (FGR). The code should include the effects of thermal conductivity degradation with burnup and should have been verified against measured fuel temperatures and stable FGR data up to the licensed burnup of the particular fuel rod design.

I-1.1 Long-lived radioactive isotopes will continue to accumulate throughout exposure, with insignificant decay because of their long half-lives. For this reason, maximum gap inventories for long-lived isotopes are likely to occur near or at the end of life of the fuel assembly.

I-1.2 Cesium is expected to behave differently from noble gases once it reaches the grain boundaries. At this point, it may react with other constituents in the fuel to form less volatile compounds that may then accumulate on the grain boundaries as solids or liquids. Cesium released from the fuel may also react with the zirconium in the cladding to form more stable (i.e., nongaseous) compounds. These effects tend to decrease the inventory of gaseous cesium available for release in the event of a cladding breach. To account for these effects, the following relationship is recommended:

$$(\text{Gap Inventory})_{\text{Cs-134, Cs-137}} = (\text{Release Fraction})_{\text{Kr-85}} * (0.5),$$

where $(\text{Gap Inventory})_{\text{Cs-134, Cs-137}}$ is the amount of gaseous cesium available for release, and $(\text{Release Fraction})_{\text{Kr-85}}$ is calculated using an approved fuel performance code.

I-2. For volatile, short-lived radioactive isotopes, such as iodine (I) (i.e., I-131, I-132, I-133, and I-135) and xenon (Xe) and Kr noble gases (except for Kr-85) (i.e., Xe-133, Xe-135, Kr-85m, Kr-87, and Kr-88), the release-to-birth (R/B) fraction should be predicted using either an NRC-approved release model or the NRC-endorsed release model from the American National Standards Institute (ANSI) / American Nuclear Society (ANS) standard ANSI/ANS-5.4, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," issued May 2011 (Ref. I-1). The prediction should use fuel parameters at several depletion time steps from an NRC-approved fuel rod thermal-mechanical performance code. The fuel parameters necessary for use in the NRC-endorsed ANSI/ANS-5.4 model calculations of the R/B fraction are local fuel temperature, fission rate, and axial node/pellet burnup. Consistent with Regulatory Position I-1, the code should include the effects of thermal conductivity degradation with burnup and should have been verified against measured fuel temperatures and stable FGR data up to the licensed burnup of the particular fuel rod design.

Because of their relatively short half-lives, the amount of activity associated with volatile radioactive isotopes depends on their rate of production (i.e., fission rate and cumulative yield), rate of release, and rate of decay. Maximum R/B ratios for short-lived isotopes are likely to occur at approximately the maximum exposure at the highest power level (i.e., the inflection point in the power operating envelope).

I-2.1 NUREG/CR-7003, "Background and Derivation of ANS-5.4 Standard Fission Product Release Model," issued January 2010 (Ref. I-2), provides guidance on using the NRC-endorsed ANSI/ANS-5.4 release model to calculate short-lived R/B fractions.

I-2.1.1 For nuclides with half-lives less than 1 hour, no gap inventories are provided. Because of their rapid decay (relative to the time for diffusion and transport), the gap fractions for these nuclides will be bounded by the calculated gap fractions for longer lived nuclides under the headings "Other Noble Gases" and "Other Halogens."

I-2.1.2 For nuclides with half-lives less than 6 hours, an approved fuel performance code is applied to predict the R/B fraction using equation 12 in NUREG/CR-7003 and its definitions of terms, as follows:

$$\left(\frac{R}{B}\right)_{i,m} = \left(\frac{S}{V}\right)_{i,m} \sqrt{\frac{\alpha_{nuclide} D_{i,m}}{\lambda_{nuclide}}}$$

where

R is the release rate (atoms per cubic centimeter per second (atoms/(cm^3-s))),
B is the production rate (atoms/(cm^3-s)),
S is surface area (cm^2),
V is volume (cm^3),
α accounts for precursor enhancement effects and is defined below in Table I-1,
D is the diffusion coefficient (cm^2/s), and
λ is the half-life (s^{-1}).

I-2.1.3 For nuclides with half-lives greater than 6 hours, the R/B fraction is predicted by multiplying the fractal scaling factor (F_{nuclide}) by the predicted Kr-85m R/B fraction using equation 13 of NUREG/CR-7003, as follows:

$$\left(\frac{R}{B}\right)_{i,nuclide} = F_{nuclide} \left(\frac{S}{V}\right)_{i} \sqrt{\frac{\alpha_{Kr-85m} D_{i}}{\lambda_{Kr-85m}}} \; .$$

The R/B fraction for the isotope I-132 should be calculated using this equation even though its half-life is less than 6 hours (2.28 hours), because its precursor of tellurium (Te)-132 has a half-life of 3.2 days, which controls the release of I-132.

I-2.1.4 Table I-1 lists the fractal scaling factors for each nuclide, calculated using the following equation from NUREG/CR-7003:

$$F_{nuclide} = \left(\frac{\alpha_{nuclide} \lambda_{Kr-85m}}{\lambda_{nuclide} \alpha_{Kr-85m}}\right)^{0.25} \; .$$

Table I-1. Fractal Scaling Factors for Short-Lived Nuclides

NUCLIDE	NUREG/CR-7003, TABLE 1			FRACTAL SCALING FACTOR
	Half-Life	Decay Constants (1/sec)	Alpha α	
Xe-133	5.243 days	1.53×10^{-6}	1.25	2.276
Xe-135	9.10 hours	2.12×10^{-5}	1.85	1.301
Xe-135m	15.3 minutes	7.55×10^{-4}	23.50	1.005
Xe-137	3.82 minutes	3.02×10^{-3}	1.07	0.328
Xe-138	14.1 minutes	8.19×10^{-4}	1.00	0.447
Xe-139	39.7 seconds	1.75×10^{-2}	1.00	0.208
Kr-85m	4.48 hours	4.30×10^{-5}	1.31	1.000
Kr-87	1.27 hours	1.52×10^{-4}	1.25	0.721
Kr-88	2.84 hours	6.78×10^{-5}	1.03	0.840
Kr-89	3.15 minutes	3.35×10^{-3}	1.21	0.330
Kr-90	32.3 seconds	2.15×10^{-2}	1.11	0.203
I-131	8.04 days	9.98×10^{-7}	1.00	2.395
I-132	2.28 hours	8.44×10^{-5}	137*	2.702
I-133	20.8 hours	9.26×10^{-6}	1.21	1.439
I-134	52.6 minutes	2.20×10^{-4}	4.40	0.900
I-135	6.57 hours	2.93×10^{-5}	1.00	1.029

* The I-132 alpha term accounts for the significant contribution from precursor Te-132.

I-3. High-confidence upper tolerance release fractions should be calculated using an NRC-approved fuel rod thermal-mechanical code, along with quantified model uncertainties.

I-3.1 For short-lived isotopes, the 2011 release model standard ANSI/ANS-5.4 recommends multiplying the best-estimate predictions by a factor of 5.0 to obtain upper tolerance release fractions.

I-3.2 For long-lived isotopes, established model uncertainties associated with the NRC-approved fuel rod thermal-mechanical code should be applied, either deterministically or sampled within a statistical application methodology, to obtain high-confidence upper tolerance release fractions.

I-4. Nominal fuel design specifications (excluding tolerances) may be used.

I-5. Actual in-reactor fuel rod power histories may diverge from reload core depletion calculations because of unplanned shutdowns or power maneuvering. Therefore, the rod power history or histories used to predict gap inventories should bound anticipated operation. Rod power histories from the fuel rod design analysis, based on thermal-mechanical operating limits from the core operating limits report or on radial falloff curves, should be used. The fuel rod power history used to calculate gap inventories should be verifiable.

I-5.1 The calculation supporting the bounding gap inventories in tables 3 and 4 in the main body of this guide used a segmented power history for both the boiling-water reactor and pressurized-water reactor limiting designs. Seven power histories were considered, each running at 90 percent of the bounding rod-average power, except that they ran at the linear heat generation rate limit for approximately 9 to 10 gigawatt-days per metric ton of uranium burnup (rod-average) at seven burnup intervals. Given that no single fuel rod will dominate the bounding power envelope, a segmented power history approach is an acceptable alternative to assigning fuel rod power at the maximum, burnup-dependent power level over the fuel rod lifetime.

I-6. Higher local power density (F_q) promotes more local FGR. Higher rod-average power (F_r), along with a flatter axial power distribution (F_z), promotes more FGR along the fuel stack. Sensitivity cases should be evaluated to ensure that the limiting fuel rod power history is captured.

I-7. Each fuel rod design (e.g., UO_2, $UO_2Gd_2O_3$, part-length, full-length) should be evaluated.

I-8. The minimum acceptable number of radial and axial nodes as defined in ANSI/ANS-5.4 should be used, along with the methodology of summing the release for these nodes, to determine the overall release from the fuel pellets to the fuel void volume.

APPENDIX I

REFERENCES[1]

I-1 American National Standards Institute (ANSI)/American Nuclear Society (ANS), ANSI/ANS-5.4, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," La Grange Park, Illinois, May 2011 (not publicly available in ADAMS).

I-2 U.S. Nuclear Regulatory Commission, NUREG/CR-7003, "Background and Derivation of ANS-5.4 Standard Fission Product Release Model," Washington, DC, January 2010 (ML100130186).

1 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public website at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. For problems with ADAMS, contact the Public Document Room staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC Public Document Room (PDR), where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to PDR.Resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

APPENDIX J

ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AEB	Accident Evaluation Branch
AMMD	aerodynamic mass median diameter
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
AST	alternative source term
ATF	accident tolerant fuel
Bq	becquerel(s)
BWR	boiling-water reactor
BWROG	Boiling Water Reactor Owners' Group
CEDE	committed effective dose equivalent
CFR	*Code of Federal Regulations*
Ci	curie(s)
cm	centimeter(s)
Cr	chromium
CsI	cesium iodide
DBA	design basis accident
DE	dose equivalent
DF	decontamination factor
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDE	effective dose equivalent
EDEX	effective dose equivalent from external sources
EPA	U.S. Environmental Protection Agency
EQ	environmental qualification
ESF	engineered safety feature
F	Fahrenheit
FAST	Fuel Analysis under Steady-State and Transients
FeCrAl	iron-chromium-aluminum
FF	flash fraction

FGR	fission gas release
FOIA	Freedom of Information Act
FSAR	final safety analysis report
g	gram(s)
g/cm^3	gram(s) per cubic centimeter
GDC	general design criterion/criteria
GWd/MTU	gigawatt-day(s) per metric ton uranium
I	iodine
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
Kr	krypton
kW/ft	kilowatt(s) per foot
LOCA	loss-of-coolant accident
LPZ	low-population zone
LWR	light-water reactor
μCi/g	microcurie(s) per gram
μm	micron(s)
m^3/s	cubic meter(s) per second
MD	management directive
MHA	maximum hypothetical accident
MOX	mixed-oxide fuel
MSIV	main steam isolation valve
MSLB	main steamline break
NRC	U.S. Nuclear Regulatory Commission
OMB	Office of Management and Budget
PRA	probabilistic risk assessment
psig	pounds per square inch gauge
PWR	pressurized-water reactor
R/B	release-to-birth
RADTRAD	RADionuclide Transport, Removal, and Dose Estimation
RCS	reactor coolant system
RG	regulatory guide
s	second(s)
scfh	standard cubic foot/feet per hour

SGTR	steam generator tube rupture
SRP	Standard Review Plan (NUREG-0800)
SSC	structure, system, or component
Sv	sievert(s)
Te	tellurium
TEARE	total effective aerosol removal efficiency
TEDE	total effective dose equivalent
T_{FGR}	transient fission gas release
TID	technical information document
TMI	Three Mile Island
TS	technical specification(s)
TSC	technical support center
UO_2	uranium dioxide
Xe	xenon
Zr	zirconium