U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-1389

Proposed Revision 1 to Regulatory Guide 1.183

ACRS Public Version for November 19, 2021 Meeting

Issue Date: Month 20xx
Technical Lead: Mark Blumberg
Additional Technical Contact: Paul Clifford

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 SAND-2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup of MOX Fuel," issued January 2011 (Ref. 1), and identifies significant attributes of other accident source terms that may be acceptable. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the AST. In some cases, unusual site characteristics, plant design features, or other factors may require different assumptions, which the staff will consider on an individual 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. 2), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 3), and holders of renewed licenses under 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants" (Ref. 4). In addition, although the guidance primarily reflects reviews of license amendment requests for light-water nuclear power plant licensees 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, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Applicable Regulations

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking Web site, http://www.regulations.gov, by searching for draft regulatory guide DG-1389. Alternatively, comments may be submitted to the Office of Administration, Mailstop: TWFN 7A-60M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/. The DG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML21204A065. The regulatory analysis may be found in ADAMS under Accession No. ML21204A066.

- 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 consequences analyses. A licensee who seeks to revise its current accident source term shall apply for a license amendment under 10 CFR 50.90. It also allows the NRC to issue these amendments if the applicant's analysis demonstrates with reasonable assurance that certain dose reference values are met.

    - 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," Criterion 19, "Control Room," 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 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.

    - A holder of an operating license issued before January 10, 1997, can, in accordance with 10 CFR 50.67, voluntarily revise the accident source term used in design basis radiological consequence analyses.

- 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 Part 52, Section 52.47, "Contents of Applications; Technical Information," and Section 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report," also require applicants of standard design certification and combined license applicants to provide a final safety analysis report (FSAR) that presents a safety analysis of the SSCs as a whole and to perform an evaluation of the offsite radiological consequences from accidents. 10 CFR Part 52 applicants and licensees are not required to comply with 10 CFR 50.67; however, those applicants and licensees may find that this guidance is useful for performing 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.

- A holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued before January 10, 1997, can, in accordance with 10 CFR 50.67 voluntarily revise the accident source term used in design basis radiological consequence analyses.

Related Guidance

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Ref. 5), provides guidance to the NRC staff for 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 the 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 the radiological consequence analyses for advanced 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, 52 and 54 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, 3150-0151 and 3150-0155, respectively. Send comments regarding this information collection to the Freedom of Information Act (FOIA), Library, and Information Collections Branch ((T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555 0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011, 3150-0151 and 3150-0155) Office of Management and Budget, Washington, DC, 20503.

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 the guide (Revision 1) addresses new issues identified since the guide was originally issued. These include (1) using the term maximum hypothetical accident (MHA)[1] loss-of-coolant accident (LOCA) to define the accident described 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 (ANSI/ANS) Standard 5.4, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel" (Ref. 6), (4) adding information to acknowledge the proposed RG may provide useful information for satisfying the radiological dose analysis requirements in 10 CFR Part 50 and 10 CFR Part 52 for advanced LWR design and siting, (5) providing additional guidance for modeling boiling-water reactor (BWR) main steam isolation valve (MSIV) leakage, and (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 holdup and retention of MSIV leakage within the main steam lines 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 provide 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. 7) calls for the use of PRA technology in all regulatory matters in a manner that complements the NRC'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.

In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 8), which provides estimates of the accident source term that are more physically

---

1 The maximum hypothetical accident (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. As used in this guide, the term "LOCA" refers to any accident that causes a loss of core cooling. The MHA LOCA 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.

based and that could be applied to the design of 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 the Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 1962 (Ref. 9), 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 reactors 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 (see Federal Register Notice (FRN) 64 FR 72001, December 23, 1999), and the subsequent issuance of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, issued July 2000 (Ref. 10), as implementing guidance for the rule.

A series of 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 whole body 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 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.

Revision 0 of RG 1.183 provides guidance for environmental qualification (EQ) that references the guidance in RG 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants" (Ref. 12). RG 1.89 is currently undergoing revision to incorporate guidance on an AST. Reactors licensed under 10 CFR Part 50 or applicants for licenses under 10 CFR Part 50 or 10 CFR Part 52 should use the guidance in Appendix I to Revision 0 of RG 1.183, until RG 1.89 incorporates guidance on an AST.

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. A combination of the methods contained in revision 0 or revision 1 of RG 1.183 would need additional justification.

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

[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 regulatory guide endorses a source term derived from SAND-2011-0128 and provides guidance on the acceptable attributes of other ASTs.

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 Part 50 and 52 licensees and applicants as related to the topic of this RG.

C. STAFF REGULATORY GUIDANCE

Section C, "Staff Regulatory Guidance," and the appendices of this RG contain regulatory positions that establish a method acceptable to the staff of the U.S. NRC for complying with regulations for DBA dose consequence analysis using an AST.

1. Implementation of Accident Source Term

1.1 Generic Considerations

As used in this guide, the AST is an accident source term that is derived principally from SAND-2011-0128, and 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 be used for advanced LWRs under 10 CFR Part 50 and 10 CFR Part 52 and for operating reactors under 10 CFR 50.34 and 10 CFR 50.67. This guide identifies an AST that is acceptable to the NRC staff and 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 they are compatible with maintaining a clear, logical, and consistent design basis and meet NRC regulations. The staff will approve these license amendment requests if 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 the 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 contained 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. Changes, or the net effect of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Once the staff has approved the initial AST implementation and it has become 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 final safety analysis report.

1.1.2 Defense in Depth

Licensees should evaluate the 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 modification does 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 provides 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 from some analyses based on whole body and thyroid criteria and some based on TEDE criteria. Full implementation of the 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 the 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 these 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 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 for 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 that would encompass 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.

[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.

Although the NRC based the AST provided in this guide on a limited spectrum of severe accidents, the particular characteristics are tailored specifically for DBA analysis use. 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 the insights of 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 Advanced and Passive Light-Water Reactor Applications

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. RG 1.183, Revision 1, extends the applicability of the proposed RG to use by advanced 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 RG 1.183, Revision 1 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 RG 1.183, Revision 1, as are applicable to the design.

1.2 Scope of Implementation

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

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. These are described in Regulatory Positions 1.2.1 and 1.2.2 below.

For 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, are not applicable 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 dose 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. At a minimum, for a full implementation, the MHA LOCA must be reanalyzed using the guidance in Appendix A to this guide. Regulatory Position 1.3 and Appendix A to this guide provide additional guidance on the analysis.

Since the AST and TEDE criteria would become part of the facility design basis, new applications of the AST would 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 re-evaluation 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 a clear, logical, and consistent design basis is maintained. 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 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, use of other characteristics of an AST or 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

There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of DBAs. A plant's licensing bases may include, but are 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. facility siting (10 CFR 100.11),[5] and

g. early site permits, standard design certifications, combined licenses (10 CFR Part 52).

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 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, 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 TMI-related requirements in 10 CFR 50.34, paragraphs (f)(1) through (3), except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). These requirements include the NUREG-0737 sections listed above in the bulleted list.

1.3.2 Reanalysis Guidance

Any full or selective implementation of an AST and any associated facility modification should be supported by evaluations of all significant radiological and non-radiological 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 be a function of the specific proposed facility modification[6] and 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 that analysis such that the results, or the conclusions drawn on those results, are no longer valid. The licensees 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, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.

---

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

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.

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 might 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.

For full implementation, the licensee should perform a complete MHA LOCA analysis, as described in Appendix A to this guide, at a minimum. The licensee should update other design basis analyses in accordance with the guidance in this section.

A selective implementation of an AST and any associated facility modification based on the AST should evaluate all of the radiological and non-radiological 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 implementations based on the timing characteristic of the AST (e.g., 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, evaluation of the radiological consequences and other impacts of the delay, such as blockage by debris in sump water, may be necessary. 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, AST characteristics) 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 are otherwise largely based on generic assumptions and inputs. Such cases might include post-accident 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 on the remainder of the 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 TEDE criteria. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of 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 doing so. Sensitivity analyses should avoid the inclusion of 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 characteristics of the AST and the TEDE criteria incorporated into the design basis will be addressed in all affected analyses. 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]

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 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. The licensee should update EQ analyses that have assumptions or inputs affected by the plant modification to address these impacts.

New facilities proposing to implement an AST should continue to use the guidance in Appendix I to RG 1.183, Revision 0, until RG 1.89 incorporates guidance on an AST.

1.4 Risk Implications

This guide provides regulatory assumptions that licensees should use in their calculation of 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.

---

[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.

Consideration should be given to 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 a 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. For guidance, refer to RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," issued January 2018 (Ref. 20). For additional information, refer to LIC-206, "Integrated Risk-Informed Decision-Making for Licensing Reviews," dated June 26, 2020 (Ref. 21), regarding how the staff performs integrated risk-informed decision-making for licensing reviews.

1.5 Submittal Requirements

According to 10 CFR 50.90, "Application for Amendment of License, Construction Permit, or Early Site Permit," 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. The NRC staff's finding as to whether an amendment is to be approved or rejected is partially based on the licensee's analyses, since it is these analyses that will become part of the design and licensing basis of the facility. The NRC staff performs these reviews by evaluating the information submitted in the amendment request against 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. The NRC staff's assessment may include performance of independent analyses to confirm the licensee's conclusion. NRC staff effort may be necessary to resolve critical differences in analysis assumptions, inputs, and methods used by the licensee and those deemed acceptable to the NRC staff.[8]

The amendment request should describe the licensee's analyses of the radiological and non-radiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. Licensees should ensure that adequate information, including analysis assumptions, inputs, and methods, is presented in the submittal to support the staff's assessment. 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 listing, preferably in tabular format, of all changes and associated justification being proposed between the current facility licensing basis and the requested license amendment.

If the licensee has used a current approved version of an NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used and submits the inputs that the licensee used in the calculations made with that code. In many cases, this will reduce the need for NRC staff confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.

---

[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.

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, "Applications for Nuclear Power Plants (LWR Edition)," 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 procedures described in the FSAR and 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 numeric 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 set forth 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 expended substantial effort in performing severe accident research and in developing the source terms in Sandia National Laboratories technical report SAND-2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup of MOX Fuel," issued January 2011. However, this RG is not a full endorsement of SAND-2011-0128. Regulatory Position 3 states certain conditions and limitations for its use. However, future research may identify opportunities for changes in 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 SAND-2011-0128. An acceptable AST has the following attributes:

a. The AST is based on major accidents hypothesized for the purposes of design analyses or 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 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 representsa 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, be verified and validated, and be 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 J to this guide provides an acceptable analytical procedure for calculating plant-specific or fuel rod design-specific fission product release fractions. U.S. NRC, Internal Memorandum, "Technical Bases for Draft RG 1.183 Revision 1 (2021) Non-LOCA Fission Product Release Fractions," (Ref. 24) provides an example calculation illustrating the application of this analytical procedure.

3. Accident Source Term

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.

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, current licensed values for fuel enrichment, 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. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values.[9] The core inventory should be determined using an appropriate isotope generation and depletion computer code. Core inventory factors (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 and 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 of the damaged fuel rods 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

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

should apply the radial peaking factors from 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[10]

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

Tables 1 and 2, below are hybridized accident source terms from SAND-2011-0128 that utilize the maximum release fractions from low- and high burnups results. This approach accounts for different radionuclide quantities at different burnups throughout the operating cycle. "Applicability of Source Term for Accident Tolerant Fuel, High-Burnup and Extended Enrichment," (Ref. 25) in part, documents the applicability of SAND-2011-0128 for burnups up to 68 gigawatt-days per metric ton of uranium (GWd/MTU) peak rod average and enrichments up to 8 percent for certain near-term ATF designs which include chromium-coated cladding and chromia-doped fuel. The use of these tables are not endorsed for mixed oxide fuels and accident tolerant fuel design concepts not discussed in Ref. 25. U.S. 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," (Ref. 26) assessed the impact of fuel fragmentation, relocation, and dispersal (FFRD) behavior on the accident source terms from SAND-2011-0128. Based upon limitations described in Ref. 26, for the purposes of assessing the radiological consequences of the MHA LOCA, the impact of FFRD does not need to be considered for the range of applicability of burnups and enrichments in SAND-2011-0128.

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 groups, available for release upon cladding breach. The licensing basis of some facilities may include non-LOCA events that assume the release of the gap activity from the entire core (e.g., heavy load drop accident). 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 Tables 3 and 4 steady-state fission product release fractions is limited to currently approved full-length uranium dioxide (UO_2) fuel rod designs operating up to 68 GWd/MTU rod average burnup at power levels below the burnup-dependent power envelope depicted in Figure 1. If it can be demonstrated that local power level, rate of fission gas release, and cumulative fission gas release remain less than the limiting co-resident UO_2 fuel rod, then Table 3 and 4 steady-state fission product release fractions apply to fuel rod designs containing integral burnable absorbers (e.g., Gadolinia). If BWR part-length fuel rods are treated as full-length fuel rods with respect to overall quantity of fission products, then Table 3 steady-state fission product release fractions apply to these part-length fuel rod designs. Applicability to future fuel rod designs, including chromium (Cr)-coated zirconium (Zr) cladding, non-Zr claddings, doped UO_2 fuel, high-density fuel, and mixed-oxide fuel, will be judged on a case-by-case basis. Appendix J provides an acceptable analytical technique for calculating plant-specific or fuel rod design-specific fission product release fractions.

---

[10] The data in this section do not apply to cores containing mixed oxide fuel.

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 are provided, as follows:

pellet burnup < 50 GWd/MTU

$$T_{FGR} = \text{maximum} [(0.26 * \Delta H) - 13) / 100, 0]$$

pellet burnup ≥ 50 GWd/MTU

$$T_{FGR} = \text{maximum} [(0.26 * \Delta H) - 5) / 100, 0]$$

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 adjustments to the above empirically based correlations are needed for different radionuclides (Ref. 27). For stable, long-lived noble gases (e.g., krypton (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 (e.g., Xe-133, Kr-85m), the transient fission product release correlations should be multiplied by a factor of 0.333.

For the remaining non-LOCA DBAs which predict fuel rod cladding failure, such as 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 and should be addressed in future applications.

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. U.S. 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 provide guidance for adjusting these gap inventories for fuel rods that are predicted to experience limited fuel centerline melting.

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	2.70×10^{-3}	2.70×10^{-3}
Cerium Group	0.00	1.6×10^{-7}	1.60×10^{-7}
Lanthanides	0.00	2.00×10^{-7}	2.00×10^{-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	1.40×10^{-3}	4.0×10^{-3}	5.4×10^{-3}
Noble Metals	0.00	6.00×10^{-3}	6.00×10^{-3}
Cerium Group	0.00	1.50×10^{-7}	1.50×10^{-7}
Lanthanides	0.00	1.50×10^{-7}	1.50×10^{-7}
Molybdenum	0.00	0.10	0.10

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

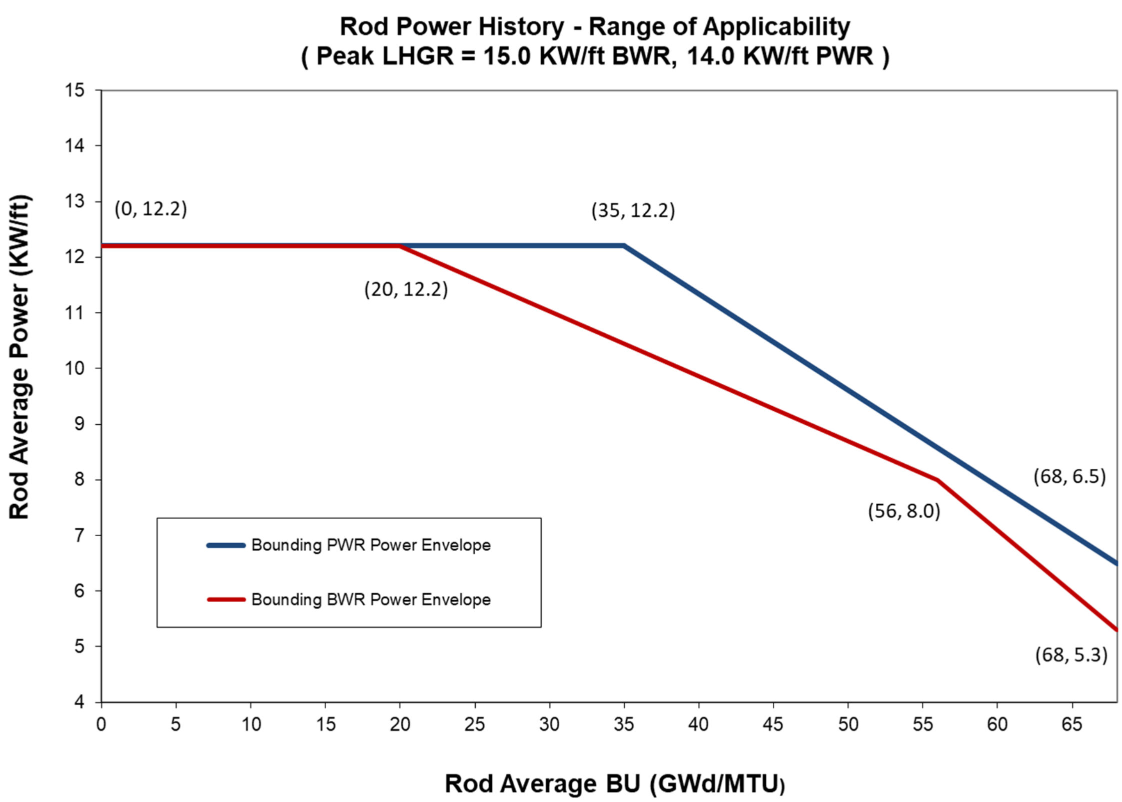


Figure 1. Maximum Allowable Power Operating Envelope for Steady-State Release Fractions

3.3 Timing of Release Phases

Table 5 provides the onset and duration 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 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	Duration	Onset	Duration
Gap Release	0.5 minutes	0.22 hours	2 minutes	0.16 hours
Early In-Vessel	0.22 hours	4.5 hours	0.16 hours	8.0 hours

The early in-vessel release phase begins immediately following the gap release phase. For facilities licensed with leak-before-break methodology, the licensee may assume the onset of the gap release phase to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the licensee should use the gap release phase onsets in Table 5. Regardless of delays in the onset, the duration of the gap release phase is 0.5 hours.

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, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium	Ce, Pu, Np

11 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).

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 provide additional details.

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

The amount of fuel damage caused by non-LOCA design basis events 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 the fuel damage caused by a fuel handling accident.

4. Dose Calculation Methodology

The NRC staff has determined (see, eg, Reactor Site Criteria Including Seismic and Earthquake Engineering Criteria for Nuclear Power Plants, Final Rule, 61 FR 65,157 (Dec. 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 TID-14844 calculated results. 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. It also provides guidance for the determination of 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 provided in International Commission on Radiological Protection (ICRP) Publication 30, "Limits for Intakes of Radionuclides by Workers," issued in 1979 (Ref. 28). 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. 29), provides

[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.

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. 30), 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.

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 analysis results. 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. TEDE should be determined for the most limiting receptor at the outer boundary of the LPZ for the duration of the accident and 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.

4.2 Control Room Dose Consequences

The following guidance should be used in determining the TEDE dose for demonstrating compliance with 10 CFR 50.67(b)(2)(iii). For the purpose of this RG, a transit dose is considered to be the dose that is accumulated as personnel travel to and from the control room for the duration of an accident once onsite (e.g., dose from site boundary to the control room). Licensees whose licensing basis includes transit dose should include the transit dose to demonstrate compliance with 10 CFR 50.67(b)(2)(iii). The licensee's results for the evaluation of transit dose will be evaluated on a case-by-case basis. New reactor

[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.

licensees that are required to show compliance with GDC 19 or similar control room radiological habitability principal design criteria may use this guidance.

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 to 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 Materials 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 result in nonconservative results for the control room.

4.2.3 Transport Models

The models used to transport radioactive material into and through the control room,[14] and the shielding models[15] used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to 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. Refer to Section 6.5.1, "ESF Atmosphere Cleanup Systems," of the SRP (Ref. 5) and RG 1.52,

[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. 31) and RADTRAD (Ref. 32) incorporate suitable methodologies.

[15] The nuclides used for modeling dose from airborne radioactivity inside the control room may not be conservative for determining the dose from radioactivity outside the control room.

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. 33), for guidance. 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 (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the 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.[16] For the duration of the event, the licensee should assume the breathing rate of this individual to be $3.5\text{x}10^{-4}$ m^3/s (Ref. 36).

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. 34).

Equation 1: $$EDEX_{finite} = \frac{EDEX_{\infty} V^{0.338}}{1173}$$

4.3 <u>Other Dose Consequences</u>

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 in NUREG-0737 for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I to RG 1.183, Revision 0, until RG 1.89 incorporates guidance on an AST.

---

16 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 Design Criterion 19," issued August 1974 (Ref. 34) and should not be credited twice. The ARCON96 Code (Ref. 35) does not incorporate these occupancy factors in the determination of the χ/Q values. Therefore, when using ARCON96 χ/Q values, dose calculations should include the occupancy factors.

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. 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, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 7 (e.g., a fuel handling accident). 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 in Appendix A to 10 CFR Part 50 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).

Table 7.[17] Accident Dose Criteria for EAB, LPZ, and Control Room Locations

Accident or Case	EAB and LPZ Dose Criteria (TEDE)	Control Room Dose Criteria[18] (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 Preincident 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[19] Unaffected steam generator(s): until cold shutdown is established
Fuel Damage or Preincident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Coincident Iodine Spike	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
PWR Main Steamline Break			Until cold shutdown is established
Fuel Damage or Preaccident Spike	0.25 Sv (25 rem)	0.05 Sv (5.0 rem)	
Coincident 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 cold shutdown is established
PWR Control Rod Ejection Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	30 days for containment pathway; until cold shutdown is established for secondary pathway
Fuel Handling Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	2 hours

The column labeled "Analysis Release Duration" summarizes the assumed radioactivity release durations identified in the individual appendices to this guide. Refer to these appendices for complete descriptions of the release pathways and durations.

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

18 The control room exposure period is 30 days for all accidents.

19 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.

5. Analysis Assumptions and Methodology

5.1 General Considerations

5.1.1 Analysis Quality

The analyses discussed in this guide are re-analyses 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 a specific accident sequence 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 engineered safeguards features 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 regarding 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 actuation of these features, especially for features that rely on manual intervention.

5.1.3 Assignment of Numeric Input Values

The licensee should select the numeric 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 may be 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 parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that identified in the technical specifications.[20] 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. To issue a license amendment authorizing the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to the amendment. The characteristics of the ASTs and the revised dose calculational methodology may be incompatible with many of the analysis assumptions and methods currently reflected in the facility's design basis analyses. The NRC staff may find that new or unreviewed issues are created by a particular site-specific implementation of the AST, warranting review of staff positions approved subsequent to the initial issuance of the license. However, prior design bases that are unrelated to the use of the AST, or are unaffected by the AST, may continue as 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 license-basis documents for guidance pertaining to the analysis of radiological DBAs other than those provided in this guide. The DBAs addressed in these attachments 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 should not be interpreted as indicating 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, use of staff positions inconstant with these assumptions is beyond the scope of this guidance.

The NRC is committed to using PRA insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions based on risk insights. The staff will not approve proposals that would reduce the defense in depth necessary to adequately protect the public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and

---

[20] 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 (Ref. 36), rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address possible changes in the parameter between scheduled surveillance tests.

addresses accident considerations not adequately addressed by the core damage frequency and large early release fraction surrogate indicators of overall risk.

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., TSC)[21] that the staff approved during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide, provided that such values remain relevant to the particular accident, release characteristics that affect plume rise, its release points, and receptor locations. If the previously approved values are based on a misapplication of a methodology or calculational errors are identified in the values, the NRC staff will pursue necessary corrections with the applicant or licensee. RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," issued November 1982 (Ref. 37), and the paper by Murphy and Campe (Ref. 34) document methodologies used in the past for determining χ/Q values.

RG 1.145 and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," issued June 2003 (Ref. 38), should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances or release characteristics that affect plume rise. 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 the LPZ χ/Q values are generally determined for an initial averaging period of 0–8 hours. The 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 should be assumed to occur coincident with the period of most unfavorable atmospheric 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.

---

[21] 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.

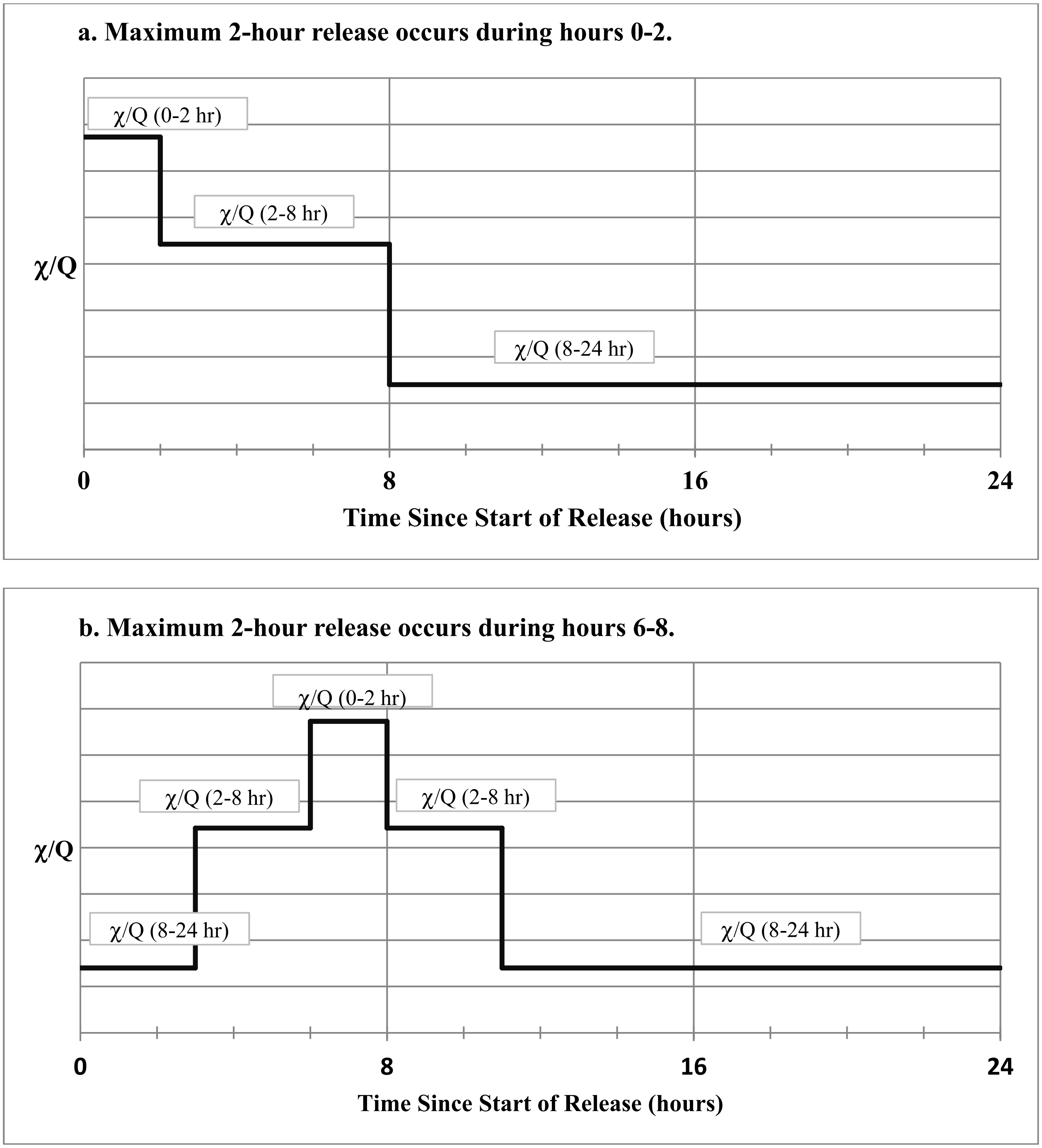


Figure 2. Example Alignments of χ/Q Values with the Maximum 2-Hour Release Period

D. IMPLEMENTATION

The NRC staff may use this RG 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 RG 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 (MD) 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 39), 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 MD 8.4. If a licensee believes that the NRC is using this RG 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 MD 8.4.

REFERENCES[1]

1. Sandia National Laboratories, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup of MOX Fuel," SAND-2011-0128, Albuquerque, NM, January 2011 (ADAMS Accession No. ML20093F003).

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

3. CFR, "Licenses, Certifications, and Approvals of Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy" (10 CFR Part 52).

4. CFR, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," Part 54, Chapter 1, Title 10, "Energy" (10 CFR Part 54).

5. U.S. Nuclear Regulatory Commission (NRC), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, Washington, DC, September 1981 (or updates of specific sections).

6. U.S. NRC, "Background and Derivation of ANS-5.4 Standard Fission Product Release Model," NUREG/CR-7003, January 2010 (ADAMS Accession No. ML100130186).

7. U.S. NRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," Federal Register, Vol. 60, August 16, 1995, p. 42622.

8. U.S. NRC, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995 (ADAMS Accession No. ML041040063).

9. U.S. Atomic Energy Commission (now U.S. Nuclear Regulatory Commission), "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 1962 (ADAMS Accession No. ML021720780).

10. U.S. NRC, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.183, Rev. 0, July 2000 (ADAMS Accession No. ML003716792).

11. CFR, "Reactor Site Criteria," Part 100, Chapter 1, Title 10, "Energy" (10 CFR Part 100).

12. U.S. NRC, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," RG 1.89, Rev. 1, June 1984 (ADAMS Accession No. ML003740271).

13. U.S. NRC, "Nuclear Regulatory Commission International Policy Statement," Federal Register, Vol. 79, No. 132, July 10, 2014, p. 39415–39418.

14. U.S. NRC, "Regulatory Guides," Management Directive 6.6, May 2, 2016 (ADAMS Accession No. ML18073A170).

---

[1] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site 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 The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

15. U.S. NRC, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396, December 1978 (ADAMS Accession No. ML051390356).

16. U.S. NRC, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654, Rev. 1 (FEMA-REP-1), November 1980 (ADAMS Accession No. ML040420012).

17. CFR, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions," Part 51, Chapter 1, Title 10, "Energy" (10 CFR Part 51)

18. U.S. NRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980 (ADAMS Accession No. ML102560051).

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

20. U.S. NRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," RG 1.174, Rev. 3 January 2018 (ADAMS Accession No. ML17317A256).

21. U.S. NRC, "Integrated Risk-Informed Decision-Making for Licensing Reviews," Office of Nuclear Reactor Regulation Office Instruction LIC-206, June 26, 2020 (ADAMS Accession No. ML19263A645).

22. U.S. NRC, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," RG 1.70, Rev. 3, November 1978 (ADAMS Accession No. ML011340122).

23. U.S. NRC, "Applications for Nuclear Power Plants," RG 1.206, Rev. 1, October 2018 (ADAMS Accession No. ML18131A181).

24. U.S. NRC Internal Memorandum, from Paul Clifford to Robert Lukes, "Technical Bases for Draft RG 1.183 Revision 1 (2021) Non-LOCA Fission Product Release Fractions," July 28, 2021 (ADAMS Accession No. ML21209A524).

25. U.S. NRC, Internal Memorandum, from Michael Case to Joseph Donoghue and Michael Franovich, "Applicability of Source Term for Accident Tolerant Fuel, High-Burnup and Extended Enrichment," May 13, 2020 (ADAMS Accession No. ML20126G376).

26. U.S. NRC, Internal Memorandum, from Hossein Esmaili to Kevin 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," July 20, 2021 (ADAMS Accession No. ML21197A067).

27. Pacific Northwest National Laboratory, "Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard," Report 18212, Rev. 1, June 2011 (ADAMS Accession No. ML112070118).

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

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

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

31. U.S. NRC, "Computer Codes for Evaluation of Control Room Habitability (HABIT V1.1)," NUREG/CR-6210, Supplement 1, October 1998 (ADAMS Accession No. ML20155K501).

32. U.S. NRC, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, April 1998 (ADAMS Accession No. ML15092A284) or NUREG/CR-7220, "SNAP/RADTRAD 4.0: Description of Models and Methods," June 2016, (ADAMS Accession No. ML16160A019).

33. U.S. NRC, "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," RG 1.52, Rev. 4, September 2012 (ADAMS Accession No. ML12159A013).

34. 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 (ADAMS Accession No. ML19116A064).

35. U.S. NRC, "Atmospheric Relative Concentrations in Building Wakes," NUREG-6331, Rev. 1, May 1997 (ADAMS Accession No. ML17213A190).

36. International Commission on Radiological Protection, "Report of Committee II on Permissible Dose for Internal Radiation," ICRP Publication 2, Pergamon Press Inc., London, 1959.

37. U.S. NRC, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," RG 1.145, Rev. 1, November 1982 (ADAMS Accession No. ML003740205).

38. U.S. NRC, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," RG 1.194, June 2003 (ADAMS Accession No. ML031530505).

---

[2] Copies of International Commission on Radiological Protection (ICRP) documents may be obtained through its Web site: 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 may be obtained through its Web site: https://www.epa.gov/libraries.

39. U.S. NRC, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Management Directive 8.4, September 2019, (ADAMS Accession No. 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, as with 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 regarding 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 (CsI), 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 pressurized-water reactors (PWRs) or the drywell in boiling-water reactors (BWRs) 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 PWRs or the drywell in BWRs 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 to 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 by 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" (the SRP) (Ref. A-2), and NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," issued July 1996 (Ref. A-3), describe acceptable models for removal of iodine and aerosols (DBA analyses should use the 10th percentile values). The analysis code RADTRAD (Ref. A-4) incorporates the latter model. The NRC staff no longer accepts the prior practice of deterministically assuming that a 50-percent plateout of iodine is released from the fuel because this value is inconsistent with the characteristics of the revised source terms. Some licensees may consider specific containment design features to evaluate aerosol fission product removal. The amount of removal will be evaluated on an individual case basis. Reduction in airborne aerosol radioactivity in the containment by both sprays and gravitational settling should be evaluated on an individual 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-5), describe acceptable models for the removal of iodine and aerosols (DBA analyses should use the 10th percentile values). The analysis code RADTRAD (Ref. A-4) incorporates this simplified model.

The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. 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 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 decontamination factor (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 aerosol 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 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," 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 Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. 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-7). 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 an individual 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 TSs.

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 reactor coolant system (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.

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 TSs. 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 TSs.

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 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-8).

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 TSs. 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 aerosol radioactivity 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 Regulatory Guide 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-7). 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 parameters 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[1] the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to Item III.D.1.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980 (Ref. A-9), would require declaring such systems inoperable. Design leakage from any systems not included in technical specifications 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 the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated and should account for the ESF leakage at accident conditions. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to the atmosphere (e.g., emergency core cooling system pump miniflow return to the refueling water storage tank).

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 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.

[1] 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.

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.[2] Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 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 to compute aerosol deposition within main steam lines. Each method computes similar removal coefficients suitable for radiological consequences calculations, however these methods are not valid if credit has been taken for aerosol removal from drywell sprays. The three MSIV leakage models are the:

1. Direct adoption of the SAND 2008-6601 (Ref. A-11) recommendations without scaling "R*-factors;"
2. Re-evaluated AEB 98-03 with multi-group; and
3. Numerical Integration.

The following assumptions 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)[3] 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 case-by-case basis. In general, the concepts used in development of 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 TSs 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

---

[2] 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-10).

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

leak rate. Section 5.4 of 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-11), describes an acceptable model for estimating the volumetric flow rate in the steamline.

A-5.4 A reduction in MSIV releases that is caused by holdup 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 (SSE) 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 are to be designed, constructed, and maintained to Quality Group A and seismic Category 1 of American Society of Mechanical Engineers (ASME) Section III requirements (Ref. A-12) or have been evaluated to be rugged as described in Regulatory Position A-5.6. The amount of reduction allowed will be evaluated on an individual 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.

In a safety evaluation dated March 3, 1999, the staff found the Boiling Water Reactor Owners Group (BWROG) technical report (TR), "Safety Evaluation of GE Topical Report, NEDC-31858P, Revision 2, 'BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,'" an acceptable method for direct reference in individual submittals on MSIV leakage, subject to the conditions and limitations described in the safety evaluation (Ref. A-13). For the purposes of DBA radiological consequence analyses, based on the information in the BWROG TR, 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.

For non-seismic analysis of the alternative drain pathway, consistent with the BWROG TR, a detailed description of following is needed:

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 path;

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 path to the condenser;

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; and

f. assurance that the valves required to open the drain path to the condenser are included in the plant's inservice testing program.

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. Also, licensees that have not previously applied for such approval may do so in accordance with the guidance in Reference A-13 or using the revised seismic analysis method described below (derived from Reference A-14) for establishing a seismically rugged alternative drain path to the condenser. Licensees choosing either of these methods should define the alternative drain pathway to the main condenser and 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 path as described in Regulatory Position A-5.4.

Note: The revised 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.

Revised Seismic Analysis of the Alternative Drain Pathway

All licensees choosing this alternative should describe the code of record used for the main steam lines and the extent of quality assurance measures applied to the design, materials, and fabrication of the steam lines and attached piping. The description should also include the alternate pathway identified to the main condenser and the basis for reliability of this pathway. In addition, the following information should be provided 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., ASME B31.1, "Power Piping" (Ref. A-15)), and the magnitude of the seismic response spectrum for the analysis equals or exceeds the licensee's SSE, 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, "Power Piping") 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 less than 0.4g, the justification should include (1) discussion of seismic capacity and margin present in the relevant SSCs, including the condenser, based on their design code(s) of record, (2) insights from plant-specific seismic assessment performed as part of the Individual Plant Examination for External Events for relevant SSCs, (3) a walkdown of a sample of the relevant SSCs, including the condenser, performed by knowledgeable licensee staff to verify that they have been constructed as designed, and (4) 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 applied to design and fabrication. Details of the walkdown(s), including qualification of licensee staff 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, "Power Piping") 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, the justification should include (1) discussion of seismic capacity and margin present in relevant SSCs, including the condenser, based on their design code(s) of record, (2) Individual Plant Examination for External Events insights described above, (3) walkdown(s) of the SSCs in the alternate pathway including the condenser, performed by knowledgeable licensee staff 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 (4) 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 applied to design and fabrication. Details of the walkdown(s), including qualification of licensee staff performing them, should be retained in archival documentation.

A-5.6 Aerosol deposition in horizontal volumes that meet Regulatory Positions A-5.4 or A-5.5 may be credited as described below.[4] For BWR designs other than plants with Mark I, II, or III containment design, other models of aerosol deposition will be considered on a case-by-case basis.

A-5.6.1 SAND 2008-6601 Model: Section 6.4 of Reference A-11 describes an acceptable model for estimating the aerosol deposition between closed MSIVs and downstream of the MSIVs. Table A-1 below provides the removal coefficients recommended by Reference A-11 and given in Table 6-1.

Table A-1. BWR Main Steamline and Condenser Removal Coefficients (hr^{-1})

Time (hr)	**Inboard**	**Between MSIVs**	**Outboard**	**Condenser**
0 - 10	0.0	1.8	1.0	0.015
10+	0.0	1.0	0.7	0.012

A-5.6.2 Re-Evaluated AEB 98-03 with 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's (Ref. A-17) use of the Stokes settling velocity physics parameters with the multi-group method which will be evaluated on an individual case-by-case basis. 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, utilize 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, volume equivalent diameter do not need to be defined.

[4] The credit described in this regulatory position will supersede the aerosol settling estimates previously provided in the NRC staff document 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-16), when Revision 1 of RG 1.183 is used.

Therefore, the Stokes settling velocity can be re-written in terms of the aerodynamic diameter, d_a as:

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

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
μ = viscosity

The State-of-the-Art Report [SOAR] on Nuclear Aerosols (Ref. A-17) 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 SOAR recommends a log-normal distribution for aerosols in the reactor coolant system (AMMD 1.0 μm with a geometric standard deviation, σ_g, of 2.0), and provides PHÉBUS-FP 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 containment aerosol (i.e., considers the effects of an active emergency core cooling system) the methods in Appendix A, Regulatory Positions 5.6.2 and 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 viscosity of 1.93x10^{-5} Pa-sec. At least 10,000 trials are necessary to develop a settling velocity distribution dataset. Note that while methods discussed in Regulatory Positions A-5.6.2 and A-5.6.3 are, in part, discussed in Reference A-19, Reference A-19 does not establish regulatory positions. For example, regarding Reference A-19, input parameters such as the AMMD, assumed in example calculations and statements regarding the validity of the existing 20-group method are not endorsed in this regulatory guidance.

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 relative probabilities for 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 mid-point of each group to represent the settling velocity for that group.

b. Compute each group, aerosol filter efficiency using the following method. By rearranging Equations 2 and 3 from Reference A-16, the filter efficiency, η_{filt}, is computed by utilizing the group settling velocity, settling area, volumetric flow rate, and 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; and,
Q = volumetric flow rate into pipe segment volume (ft^3/hr);
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-weight 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
V = well-mixed pipe free volume

A-5.6.3 Numerical Integration: The equation for a normalized number distribution, ($n(d_a)$) of particles of aerodynamic diameter (d_a) is given in Reference A-18 as:

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

where:

σ_g= geometric standard deviation
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, is by the relation:

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

Discretize the range of particle diameters, d_a, from 1E-9 to 1E-3 microns in to 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-19 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); (4) utilize Equation A-2 to then compute the filter efficiencies. Finally, utilize Equation A-3 to convert filter efficiencies, η_{filt}, into removal coefficients, hr^{-1}.

A-5.6.4 Aerosol deposition removal coefficients for the condenser using a multi-group method and numerical integration will be evaluated on an individual 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 an individual case basis. The model should assume well-mixed volumes. Reference A-20 provides guidance on an acceptable model.

A-5.8 Reduction of the amount of released organic iodine (e.g., Brockman-Bixler model in RADTRAD (Ref. A-4)) 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 above, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground-level release.

A-5.10 Holdup 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 purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The licensee may take into account the reduction in the amount of radioactive material released via ESF filter systems, provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-6).

APPENDIX A

REFERENCES

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

A-2. U.S. Nuclear Regulatory Commission (NRC), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800.[1]

A-3. U.S. NRC, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," NUREG/CR-6189, July 1996 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100130305).

A-4. U.S. NRC, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, April 1998 (ADAMS Accession No. ML15092A284); or "SNAP/RADTRAD 4.0: Description of Models and Methods," NUREG/CR-7220, June 2016 (ADAMS Accession No. ML16160A019).

A-5. U.S. NRC, "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, June 1993 (ADAMS Accession No. ML063480542).

A-6. U.S. NRC, "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," Regulatory Guide 1.52, Rev. 4, September 2012 (ADAMS Accession No. ML12159A013).

A-7. U.S. NRC, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Information Notice 91-56, September 19, 1991.

A-8. U.S. NRC, "Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control," Information Notice 88-76, September 19, 1988.

A-9. U.S. NRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980 (ADAMS Accession No. ML102560051).

A-10. U.S NRC, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995 (ADAMS Accession No. ML041040063).

A-11. Sandia National Laboratories, "Analysis of Main Steam Isolation Valve Leakage in Design Basis Accidents Using MELCOR 1.8.6 and RADTRAD," SAND2008-6601, Letter Report, Albuquerque, NM, October 2008 (ADAMS Accession Nos. ML083180196 and ML113400138 (Errata)).

---

1 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site 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 The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

A-12. American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, “Nuclear Power Plant Components,” New York, NY.[2]

A-13. U.S. NRC, Letter, “Safety Evaluation of GE Topical Report, NEDC-31858P [Proprietary GE Report], Revision 2, ‘BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems,’ September 1993,” March 3, 1999 (ADAMS Accession No. ML010640286).

A-14. U.S. NRC, “Draft Technical Assessment of Hold-up and Retention of Main Steam Isolation Valve Leakage Within the Main Steam Lines and Main Condenser,” (ADAMS Accession No. ML20085J042).

A-15. American Society of Mechanical Engineers, B31.1, “Power Piping,” Issued 2001, New York, NY.

A-16. U.S. NRC, “Assessment of Radiological Consequences for the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term,” Report AEB-98-03, December 9, 1998 (ADAMS Accession No. ML011230531).

A-17. Allelein, H.-J. A. (2009). Technical Report NEA/CSNI/R(2009)5, State-of-the-Art Report on Nuclear Aerosols, NEA/CSNI/R(2009)5. Nuclear Energy Agency / Committee on the Safety of Nuclear Installations (NEA/CSNI).

A-18. Williams, M.M.R. (1991). Aerosol Science Theory and Practice, New York: Pergamon Press.

A-19. U.S. NRC, Internal Memorandum from Elijah Dickson to Kevin Hsueh, “Technical Bases 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.” July 2021. (ADAMS Accession No. ML21141A006).

A-20. J.E. Cline & Associates, Inc., “MSIV Leakage Iodine Transport Analysis,” Letter Report, March 26, 1991 (ADAMS Accession No. ML003683718).

2 Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Two Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web site store at http://www.asme.org.

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 light-water reactors. These assumptions supplement the guidance in the main body of this guide.[1]

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 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., 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 utilize, in part, the transport model discussed in Reference B-1 and those contained in Reference 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. Elemental iodine and organic iodine are conservatively assumed to be in vapor form and subsequently decontaminated by passage through the overlying pool of water into the building atmosphere.

Phase 2—the protected release of CsI re-evolving as elemental iodine. CsI is conservatively assumed to completely dissociate into the pool water. Because of the low pH of the pool water, CsI (and Phase 1 absorbed elemental iodine and organic iodine within the pool) then slowly re-evolve as elemental iodine into the building atmosphere.

1 These assumptions may also be used in assessing the radiological consequences of a heavy load drop over fuel accident. If the event is postulated to damage all of the rods in the core, the release activity may be based on the core-average gap fractions of Tables 1 and 2 in the main text of the guide, and the radial peaking factor may be omitted.

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

An overall 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, an overall effective iodine DF for elemental iodine and organic 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 less decay power. The limiting pool water temperature near fuel rods and basing the estimate on a full-core offload may be used to determine internal gas temperature and thus pin pressure.

For water depths between 19 and 23 feet, an overall iodine DF based on pin pressure is computed as follows:

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

where:

t = bubble rise time (sec), computed as a function of pin pressure, x (psig), as:

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

d = bubble diameter (cm), computed as a function of pin pressure, x (psig), as:

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

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.

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

The re-evolution calculation results in a simple exact transient solution. It has the flexibility to consider the effect of potential filtration and other removal mechanisms. The following information is needed:

Site-specific and general parameters:

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. $N_{I\text{-}129gap}$—fuel pin nonradioactive iodine in gap (moles),

f. K_L = mass transfer coefficient—3.66×10^{-6} m/s [2], 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. (For the purpose of calculating concentrations in M (moles/liter), V_{pool} must be converted to liters.)

Calculation Sequence:

1. Calculate amount of iodine (radioactive and nonradioactive) in the fuel pin gap.
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.

Step 1—Calculate amount of iodine in the fuel pin gap.

Both the radioactive and nonradioactive iodine in the pool affect the radioactive iodine evolution. The calculations operate on moles so iodine isotope quantities must be converted to moles.

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:

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

$$N_{I-129} = \left(\frac{m_{I-129}(g)}{M_{I-129}\left({}^{g}/_{mol}\right)}\right) \quad \text{(Equation B-5)}$$

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

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

Activities in curies (Ci) must be converted to becquerel (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.

Step 2—Calculate volatile iodine fraction in pool.

Next, determine the fraction of iodine atoms in the pool that are in I_2 (volatile) form:

2 See NRC, "Re-evaluation of the Fission Product Release and Transport for the Design-Basis Accident Fuel Handling Accident," E. Dickson, M. Salay, issued December 2019 (Ref. B-1).

- Calculate the radioactive and total concentrations in the pool:

  C_r = concentration (M) (moles I atoms /L) of radioactive I atoms = $N_{I\text{-}131gap} / V_{pool}$ (Equation B-7)

  C_t = total I concentration (M) (moles I atoms / L) = $(N_{I\text{-}129gap} + N_{I\text{-}131gap}) / V_{pool}$ (Equation B-8)

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

- Calculate the H+ concentration:

  $C_h = [H^+] = 10^{-pH}$ (Equation B-9)

- Calculate the $[I_2]/[I\text{-}]^2$ concentration ratio, R_i[3]((Ref. B-2):

  $R_i = [I_2]/[I\text{-}]^2 = C_h^2 / (6.0603E\text{-}14 + 1.4708E\text{-}09\ C_h)$ (Equation B-10)

- Calculate the fraction of I atoms in I_2 form:

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

    $B_m = 4\ C_t + 1 / R_i$ (Equation B-11)

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

    $X_e = (B_m - \sqrt{B_m^2 - 16\ C_t^2}\ ) / (4\ C_t)$ (Equation B-12)

Step 3—Calculate applicable removal coefficients.

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:

$\lambda_e = K_L\ X_e\ S_{pool} / V_{pool}$ (Equation B-13)

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.

Step 4—Evaluate release as an overall release.

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

$Q_e = \lambda_e\ V_{pool}$ (Equation B-14)

[3] Combined speciation rate from NUREG/CR-5950, "Iodine Evolution and pH Control," issued December 1992 (Ref. B-3).

a. λ_f is used if recirculation filtration is credited.
b. Alternatively, a loop and filter can be modeled 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., DF of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite DF).

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 considers time-independent releases from the fuel pool. The release rate is generally assumed to be a linear or exponential function over this time period. For the time-dependent releases from the fuel pool from the re-evolution of iodine, these releases are to be considered directly to the environment outside the fuel building as they are released from the pool.

B-5.2 A reduction in the amount of radioactive material released from the fuel pool by engineered-safety-feature (ESF) filter systems may be taken into account, provided these systems meet the guidance of 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," 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.[4]

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 Containment

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

---

4 These analyses should consider the time 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.

B-6.1 If the containment is isolated[5] 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,[2] no radiological consequences need to be analyzed for the isolated pathway.

B-6.3 If the containment is open during fuel handling operations (e.g., a personnel air lock or equipment hatch is open),[6] 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 considers time-independent releases from the reactor cavity. The release rate is generally assumed to be a linear or exponential function over this period. For time-dependent releases from the reactor cavity pool from the re-evolution of iodine, these releases are to be considered directly to the environment outside the containment as they are released from the pool.

B-6.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 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.[2]

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.

---

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

6 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 should generally not credit this manual isolation.

APPENDIX B

REFERENCES

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," Staff Technical Paper, 2019. (ADAMS Accession No. ML19248C647).

B-2. US. NRC, Staff Technical Paper, "Internal Memorandum, from Elijah Dickson to Kevin Hsueh, "Example Calculation of Re-Evaluated Fuel Handing Accident Fission Product Transport Model for Draft RG 1.183 Revision 1 (2021)." August 2021. (ADAMS Accession No. ML21190A040).

B-3. U.S. NRC, "Iodine Evolution and pH Control," NUREG/CR-5950, Oak Ridge National Laboratory, December 1992 (ADAMS) Accession No. ML063460464).

B-4. U.S. NRC, "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," Regulatory, Guide 1.52, Revision 4, September 2012. (ADAMS Accession No. ML12159A013).

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 boiling-water reactors. These assumptions supplement the guidance provided 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.[1]

C-2. If no or minimal[2] fuel breach is postulated for the limiting event, the released activity should be the maximum coolant activity (typically a pre-accident spike of 4 microcuries/gram dose equivalent (DE) iodine (I)-131 (DE I-131)) allowed by the technical specifications.

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 either 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 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

---

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

[2] "Minimal fuel breach" is defined 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 leakage 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.

terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

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 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 accident at boiling-water reactor (BWR) light-water reactors. These assumptions supplement the guidance provided 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[1] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the standard technical specifications for the nuclear steam supply system vendor:

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

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

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 technical specifications.

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

---

[1] "Minimal fuel breach" is defined 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-4.3 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, holdup, or dilution within facility buildings.

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 pressurized-water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

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[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. 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 (typically 60 microcuries per gram (μCi/g) dose equivalent (DE) iodine (I)-131 (DE I-131)) permitted at full-power operations by the technical specifications (i.e., a 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 (typically, 1.0 μCi/g DE I-131) specified in the technical specifications (TSs) (i.e., 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.

---

[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998 (Ref. E-1), for acceptable assumptions and methodologies for performing radiological analyses.

[2] "Minimal fuel breach" is defined 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-4. The specific activity in the steam generator liquid at the onset of the SGTR is at the maximum value permitted by secondary activity TSs (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

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 TSs. 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., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool 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).

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 of the leakage 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 operating and releases from the steam generator have been terminated, or the steam generator is isolated from the environment such 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 below:

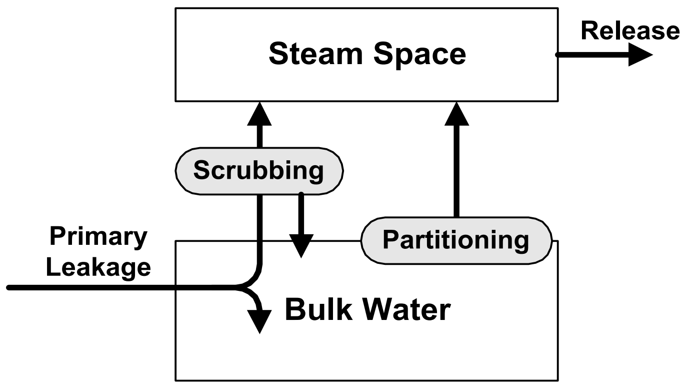


Figure E-1. Transport Model

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-2), 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 the function of the steaming rate and the partition coefficient.[3] A partition coefficient for iodine of 100 may be assumed. The retention of non-iodine particulate radionuclides in the steam generators is limited by the moisture carryover from the steam generators.

---

3 "Partition coefficient" is defined as follows:

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

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-3). 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 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 E

REFERENCES

E-1. U.S. Nuclear Regulatory Commission (NRC), "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998 (ADAMS Accession No. ML003739223).[1]

E-2. U.S. NRC, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409, January 1978 (ADAMS Accession No. ML19269F014).

E-3. U.S. NRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988 (ADAMS Accession No. ML031150151).

[1] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site 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 The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

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 accident at pressurized-water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

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[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed:

F-2.1 A reactor transient has occurred before the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 microcuries per gram (μCi/g) dose equivalent (DE) iodine (I)-131 (DE I-131)) permitted by the technical specifications (TSs) (i.e., a 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 (typically, 1.0 μCi/g DE I-131) specified in the TSs (i.e., 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.

---

1 Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998 (Ref. F-1), for acceptable assumptions and methodologies for performing radiological analyses.

2 "Minimal fuel breach" is defined 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.

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 TSs (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 to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

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[3] 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 the alternative repair criteria (ARC) (see Ref. F-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. For facilities with traditional steam generator specifications (both per generator and 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 1.9×10^3 liters per day (lpd) (500 gallons per day (gpd)) 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 1.9×10^3 lpd (500 gpd) to the faulted generator and 1.2×10^3 lpd (313 gpd) to each of the unaffected generators.

For facilities that have implemented ARC, 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 TSs is equally apportioned between the unaffected steam generators.

F-6.3 The density used in converting volumetric leak rates (e.g., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications 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 of the leakage is less than 100 degrees Celsius (212 degrees Fahrenheit). 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.

3 "Faulted" refers to the state of the steam generator in which the secondary side has been depressurized by an MSLB such that protective system response (main steamline isolation, reactor trip, safety injection, etc.) has occurred.

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.

F-6.6.1 With regard to 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 the function of the steaming rate and the partition coefficient. A partition coefficient[4] for iodine of 100 may be assumed. 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-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 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.

4 "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 F

REFERENCES

F-1. U.S. Nuclear Regulatory Commission (NRC), "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998 (ADAMS Accession No. ML003739223).[1]

F-2. U.S. NRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988 (ADAMS Accession No. ML031150151).

[1] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site 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 The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

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 pressurized-water reactors.[1] These assumptions supplement the guidance provided 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 and 3-percent organic. These fractions apply 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. 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., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool 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).

---

[1] Facilities licensed with, or applying for, alternative repair criteria should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998, for acceptable assumptions and methodologies for performing radiological analyses.

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 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 pressurized-water reactors.[1] These assumptions supplement the guidance provided 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.[2]

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 accident loss-of-coolant accident, main steamline break, and steam generator tube rupture.

H-3. In the first release case, 100 percent of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second case, 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 accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual 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.

---

[1] Facilities licensed with, or applying for, alternative repair criteria should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," issued December 1998, for acceptable assumptions and methodologies for performing radiological analyses.

[2] Calculated values of the combined release (gap activity plus fuel melt) are limited to a value of 1.0.

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.

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 (TSs) 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 TSs 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 TSs.

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 TSs should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.

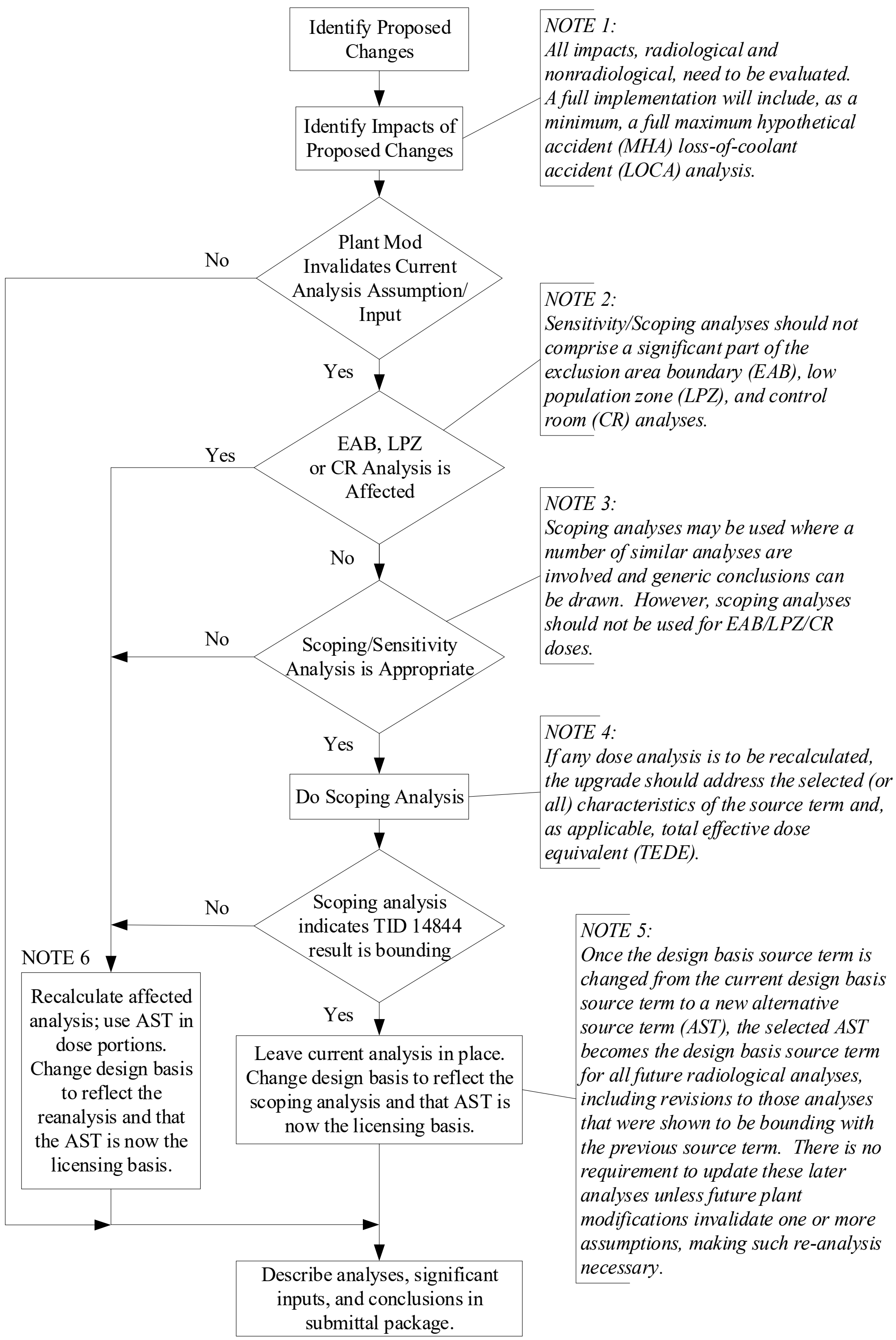
H-7.2 The density used in converting volumetric leak rates (e.g., gallons per minute) to mass leak rates (e.g., pound mass per hour) should be consistent with the basis of surveillance tests used to show compliance with leak rate TSs. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is 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

ANALYSIS DECISION FLOWCHART



APPENDIX J

ANALYTICAL TECHNIQUE FOR CALCULATING FUEL-DESIGN OR PLANT-SPECIFIC STEADY-STATE FISSION PRODUCT RELEASE FRACTIONS FOR NON-LOCA 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 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 Section C of 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 design).

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 attachment prescribes 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.

The U.S. Nuclear Regulatory Commission (NRC) maintains 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-approved 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 include the following attributes:

J-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.

J-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.

J-1.2 Cesium is expected to behave differently than noble gas 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., non-gaseous) 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.

J-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), an NRC-approved release model or an NRC-endorsed ANS-5.4 release model should be used to predict the release-to-birth (R/B) fraction using 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 ANS-5.4 model calculations of the R/B fraction are local fuel temperature, fission rate, and axial node/pellet burnup. Consistent with Item J-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., inflection point in the power operating envelope).

J-2.1 NUREG/CR-7003 (PNNL-18490), "Background and Derivation of ANS-5.4 Standard Fission Product Release Model," issued January 2010 (ADAMS Accession No. ML100130186), provides guidance on using the NRC-endorsed ANS-5.4 release model to calculate short-lived R/B factors.

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

J-2.1.2 For nuclides with half-lives of 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 release rate (atoms-cm^{-3}s^{-1}),
B is production rate (atoms-cm^{-3}s^{-1}),
S is surface area (cm^{2}),
V is volume (cm^{3}),
α accounts for precursor enhancement effects and is defined below,

D is diffusion coefficient (cm^2s^{-1}), and
λ is half-life (s^{-1}).).

J-2.1.3 For nuclides with half-lives of 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 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 isotope I-132 should be calculated using this fractal 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.

J-2.1.4 Table J2 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 J-2. Fractal Scaling Factors for Short-Lived Nuclides

NUCLIDE	NUREG/CR-7003, TABLE 1			FRACTAL SCALING FACTOR
	Half-Life	Decay Constants	Alpha	
Xe-133	5.243 days	$1.53x10^{-6}$	1.25	2.276
Xe-135	9.10 hours	$2.12x10^{-5}$	1.85	1.301
Xe-135m	15.3 months	$7.55x10^{-4}$	23.50	1.005
Xe-137	3.82 months	$3.02x10^{-3}$	1.07	0.328
Xe-138	14.1 months	$8.19x10^{-4}$	1.00	0.447
Xe-139	39.7 seconds	$1.75x10^{-2}$	1.00	0.208
Kr-85m	4.48 hours	$4.30x10^{-5}$	1.31	1.000
Kr-87	1.27 hours	$1.52x10^{-4}$	1.25	0.721
Kr-88	2.84 hours	$6.78x10^{-5}$	1.03	0.840
Kr-89	3.15 months	$3.35x10^{-3}$	1.21	0.330
Kr-90	32.3 seconds	$2.15x10^{-2}$	1.11	0.203
I-131	8.04 days	$9.98x10^{-7}$	1.00	2.395
I-132	2.28 hours	$8.44x10^{-5}$	137*	2.702
I-133	20.8 hours	$9.26x10^{-6}$	1.21	1.439
I-134	52.6 months	$2.20x10^{-4}$	4.40	0.900
I-135	6.57 hours	$2.93 x10^{-5}$	1.00	1.029

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

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

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

J-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.

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

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

J-5.1 The calculation supporting the bounding gap inventories in Table J-1 used a segmented power history (SPH) for both the boiling-water reactor and pressurized-water reactor limiting designs. Seven different power histories were considered, with each running at 90 percent of the bounding rod average power, with the exception of running at the linear heat generation rate limit for approximately 9 to 10 gigawatt-days per metric ton of uranium (GWd/MTU) burnup (rod average) at seven different burnup intervals. Given that no single fuel rod will dominate the bounding power envelope, a SPH approach is an acceptable alternative to assigning fuel rod power at the maximum, burnup-dependent power level over the fuel rod lifetime..

J-6. Higher local power density (Fq) promotes more local FGR. Higher rod average power (Fr), along with a flatter axial power distribution (Fz), promotes more FGR along the fuel stack. Sensitivity cases should be conducted to ensure the limiting fuel rod power history is captured.

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

J-8. The minimum acceptable number of radial and axial nodes as defined in 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 K

ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ANSI/ANS	American National Standards Institute/American Nuclear Society
ARC	alternative repair criteria
ASME	American Society of Mechanical Engineers
AST	alternative source term
BU	burnup
Bq	becquerel
BWR	boiling-water reactor
BWROG	Boiling-Water Reactor Owners Group
CFR	*Code of Federal Regulations*
CEDE	committed effective dose equivalent
Ci	curie
CR	control room
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
FF	flash fraction
FGR	fission gas release
FSAR	final safety analysis report
GDC	general design criterion/criteria
GWd/MTU	gigawatt day per metric ton of uranium

gpd	gallon(s) per day
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
KW/ft	kilowatts per foot
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LPZ	low population zone
LWR	light-water reactor
$\mu Ci/g$	microcuries per gram
m^3/s	cubic meters per second
MHA	maximum hypothetical accident
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
PWR	pressurized-water reactor
R/B	release-to-birth
RADTRAD	RADionuclide, Transport, Removal and Dose Estimation
RCS	reactor coolant system
RG	regulatory guide
RM	radiation monitor
scfh	standard cubic feet per hour
SGTR	steam generator tube rupture
SRP	Standard Review Plan
SSC	structure, system, and component
Sv	sievert(s)
TEARE	total effective aerosol removal efficiency
TEDE	total effective dose equivalent
TID	technical information document
TR	technical report
TSC	technical support center
UFSAR	updated final safety analysis report
UO_2	uranium dioxide