C. STAFF REGULATORY GUIDANCE

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

This RG applies to MHA LOCA models for applicants and licensees using zirconium-alloy cladded uranium dioxide (UO_2) fuel rod designs with reactor core burnups up to a maximum rod-average of ~~68~~80 gigawatt-days per metric ton uranium (GWd/MTU) (and fuel enrichments up to 8 weight-percent uranium-235~~), including chromium-coated cladding (thicknesses less than 50 microns (μm))~~ for PWRs and ~~chromia-doped (up to 0.16~~10 weight-percent for BWRs), for currently approved (as of issuance of this RG revision) fuel~~–~~ and cladding materials, examples of which includes coated zirconium alloy claddings and doped fuels. It is ~~not~~also applicable for ~~other fuel-clad combinations, including~~ fuel with either iron-chromium-aluminum (FeCrAl) alloy and chromium coated cladding~~.~~ (Ref XX).

This RG applies to non-LOCA models for applicants and licensees with reactor core burnups up to a maximum rod-average burnup of ~~68~~80 GWd/MTU (and fuel enrichments ~~up to~~of 8 weight-percent uranium-235 for PWRs and 10 weight-percent for BWRs) for currently approved (as of the issuance of this RG) zirconium-alloy cladded UO_2 fuel rod designs at power levels below the burnup-dependent power envelopes depicted in figure 1 of this guide. For rod designs outside of these limits, Appendix I provides an acceptable technique for calculating maximum steady-state release fractions.

1. Implementation of Accident Source Term

1.1 Generic Considerations

As used in this guide, the AST is an accident source term derived principally from ~~SAND2011-0128~~SAND2023-01313; it differs from the SAND2011-0128, TID-14844, and NUREG-1465 source terms used in the original and revised design and licensing of operating reactor facilities. ASTs may also be used by applicants for new LWRs, including advanced evolutionary and passive LWRs, under 10 CFR Part 50 and 10 CFR Part 52, and for existing operating reactor licensees under 10 CFR 50.34 and 10 CFR 50.67. This guide identifies an AST that is acceptable to the NRC staff and describes significant characteristics of other source terms that may be found acceptable. While the staff recognizes several potential uses of an AST, it is not possible to foresee all possible uses. Licensees may pursue technically justifiable uses of the ASTs in the most flexible manner in license amendments so long as these uses are compatible with maintaining a clear, logical, and consistent design basis and continue to comply with NRC regulations. These license amendment requests should demonstrate that the facility, as modified, will continue to provide sufficient safety margins, with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs.

1.1.1 Evaluation of Defense in Depth and Safety Margins

One aspect of the engineering evaluation is to show that the proposed change does not compromise the fundamental safety principles that are the basis of plant design and operation (i.e., activities such as maintenance, testing, inspection, and qualification).[KM-02] During the design process, plant response and associated safety margins are evaluated using assumptions of physical properties and operating characteristics. National standards, the defense-in-depth philosophy, and the General Design the fundamental principles and elements of evaluating defense-in-depth and safety margin. (Ref. XX) For consistency between regulatory documents, these principles and elements are also discussed in the context of license amendment requests pertinent to design basis accident radiological consequence analyses.

A licensee's proposed change might affect safety margins and defenses incorporated into the current plant design and operation; therefore, the licensee should reevaluate the safety margins and layers of defense to support the proposed change. As part of this evaluation, the licensee should determine the impact of the proposed licensing basis change on the functional capability, reliability, and availability of affected equipment. The plant's licensing basis is the reference point for judging whether a proposed change adversely affects safety margins or defense in depth. Sections 1.1.1.1 and 1.1.1.2 present guidance on assessing whether the proposed change maintains adequate safety margins and remains consistent with the defense-in-depth philosophy.

1.1.1.1 Safety Margins

This regulatory guide's methodology of radiological consequence analyses uses a system of coupled, first-order differential equations to describe the behavior of the source term traveling from the reactor core, through the containment, and other plant systems, to the environment and eventually to a dose receptor. Mathematically, the methodology is straight forward and conservative, intended to be readily understood by a reviewer and bounding the complexities of reality for an engineered system. The models can be simplistic to very complex, ranging from a few dozen to several hundred input parameters. These input parameters not only model the design of the engineered system itself, but other aspects as well, such as meteorological conditions, human performance, and health effects of exposure to ionizing radiation, various physical processes to mitigate the source term, as well as representative testing data. As such, the uncertainties in these models are quite large and results are typically presented as point estimates (i.e., without uncertainty characterizations).

A significant consideration in these models is the inclusion of "safety factors" placed on input parameters to maintain a conservative model and account for uncertainty. The need for conservative biasing arises from the inherent uncertainties that exist in real-world engineering applications. These uncertainties can stem from various sources, such as variations in environmental conditions, operator actions, unexpected system failures, or unforeseen events. Safety factors help account for these uncertainties by introducing a layer of conservatism into the analysis results. This conservatism ensures that the system not only meet their intended functions but also maintains a high level of reliability to withstand conditions that may exceed original design expectations.

Licensees should evaluate their proposed uses of this guide, and the associated proposed facility modifications and changes to procedures, to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times. For example, changes, or the net effect of multiple changes, that result in a reduction in safety margins may require prior NRC approval. If the initial AST implementation, consistent with the guidance in RG 1.183, Revision ~~1~~2, is approved by the staff and becomes part of the facility design basis, licensees may use 10 CFR 50.59, "Changes, tests and experiments," and its supporting guidance to assess facility modifications and changes to procedures that are described in the updated FSAR.

Considering safety margin is consistent with the expectations for showing compliance with regulations. When the computed radiological consequence results are close to the established acceptance criteria, a margin analysis can be a helpful step in assessing the level of conservatism in the model. For instance, licensees can systematically evaluate how the variations or sensitivities in select input parameters, and associated safety factors, can lead to results that fall either above or below the established acceptance criteria. Additionally, licensees can assess the analytical margin included in the model by increasing the model fidelity. This is done by more realistically assessing simplifying modeling assumptions of highly complex engineered systems. For example, modeling the actual number of main steam lines vice using two lines—one representing the line with a failed isolation valve and the other representing the multiple lines that do not have failed components. Either approach is a valuable exercise as it helps quantify the model's robustness and sensitivity to key variables while also helping to identify the amount of margin inherent to the model.

When performing margin analysis, it is acceptable for results to occasionally deviate from the established acceptance criteria, provided these deviations are within a certain acceptable range when balanced with other parameter and modeling assumption conservatisms also assumed in the model. This is because real-world systems often encounter uncertainties, changing conditions, or unexpected events that may lead to degraded performance. A margin analysis helps strike a balance between safety and efficiency, ensuring that the analysis of record is robust enough to withstand uncertainties or unexpected variations while avoiding unnecessary overdesign or detailed reanalysis. Likewise, by understanding, and quantifying these sensitivities impacting acceptable margins, decision-makers gain valuable insights into the systems behavior and can make informed choices. If a margin analysis is performed to help demonstrate compliance, a discussion of the assessed parameters, their applied safety factor with justification, and impact on the final results should be submitted for review by the staff.

1.1.1.2 Defense in Depth

Defense in depth is an element of the NRC's safety philosophy that employs successive measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense in depth philosophy has traditionally been applied in plant design and operation to provide multiple means to accomplish safety functions and prevent the release of radioactive material. It has been and continues to be an effective way to account for uncertainties in equipment and human performance and, in particular, to account for the potential for unknown and unforeseen failure mechanisms or phenomena that, because they are unknown or unforeseen, are not reflected in either the PRA or traditional engineering analyses. The SRM on SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999 (Ref. XX), provides additional information on defense in depth as an element of the NRC's safety philosophy.

Defense in depth is often characterized by varying layers of defense, each of which may represent conceptual attributes of nuclear power plant design and operation or tangible objects such as the physical barriers between fission products and the environment. The NRC implements defense in depth as four layers of defense that are a mixture of conceptual constructs and physical barriers (see NUREG/KM-0009 for further detail, Ref. XX). For the purposes of this regulatory guide, nuclear power plant defense in depth is taken to consist of layers of defense (i.e., successive measures) to protect the public:

- robust plant design to survive hazards and minimize challenges that could result in an event occurring,
- prevention of a severe accident (core damage) if an event occurs,
- containment of the source term if a severe accident occurs, and
- protection of the public from any releases of radioactive material (e.g., through siting in low-population areas and the ability to shelter or evacuate people, if necessary).

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

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

The NRC finds it acceptable for a licensee to use the following seven considerations to evaluate proposed changes that may impact defense in depth:

1. Preserve a reasonable balance among the layers of defense.
2. Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.
3. Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.
4. Preserve adequate defense against potential common cause failure.
5. Maintain multiple fission product barriers.
6. Preserve sufficient defense against human errors.
7. Continue to meet the intent of the plant's design criteria.

Regulatory Guide 1.174 presents more detail guidance on how to apply these considerations.

1.1.3 Integrity of Facility Design Basis

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

This approach for operating reactors creates two tiers of analyses—one based on the previous TID-14844 source term, and one based on an AST. The radiological acceptance criteria would also differ, as some analyses are based on whole-body and thyroid criteria, and some are based on TEDE criteria. Full implementation of an AST revises the plant licensing basis to specify the AST in place of the previous TID-14844 accident source term and establishes the TEDE dose as the new acceptance criteria. Selective implementation of an AST also revises the plant licensing basis and may establish the TEDE dose as the new acceptance criteria. Selective implementation differs from full implementation only in the scope of the change. In either case, the facility design bases should clearly indicate that the source term parameters should be examined to maximize fission product inventory. For non-LOCA DBAs, the NRC staff will consider on a case-by-case basis the use of more explicit methods to calculate the fission product inventory and reactor coolant system activity, such as using the burnup, power history, and peaking factor from the accident analyses (e.g., chapter 14 or 15) in the updated FSAR, for each rod predicted to fail. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach maximum values.[13] The core inventory should be determined using an appropriate computer code for calculating isotope generation and depletion. The core inventory factors (in curies per megawatt thermal) provided in TID-14844 and used in some analysis computer codes were derived for low-burnup, low-enrichment fuel and should not be used with higher burnup or higher enrichment fuels. The code should model the fuel geometries, material composition, and burnup, and the cross section libraries used should be applicable to the projected fuel burnup.

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

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

3.2 <u>Release Fractions</u>

For the MHA LOCA, table 1[14] (for BWRs) and table 2 (for PWRs) in this RG list the core inventory release fractions, by radionuclide group, for the gap release and early in-vessel damage phases. These fractions are applied to the maximum core inventory described in Regulatory Position 3.1.

The limitations on the use of tables 1 and 2 in this RG are based on several reference documents. First, tables 1 and 2 are based, in part, on accident source terms from ~~SAND2011-0128~~SAND2023-01313 that use the maximum release fractions from low- and high-burnup results. ~~The tables 1 and 2~~[KM-04] These source terms were derived by examining a set of accident sequences for current LWR designs; they reflect the current understanding of severe accidents and fission product behavior since the publication of NUREG-1465. The use of the maximum release fractions from low- and high-burnup results accounts for different radionuclide quantities at different burnups throughout the operating cycle. Second, the NRC internal memorandum "Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment," dated May 13, 2020 (Ref. 25), in part, and insights from a literature review on ATFs reported in NUREG/CR-7282, "Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases," issued July 2021 (Ref. 26), support the applicability of ~~SAND2011-0128~~SAND2023-01313 for licensees using zirconium-alloy cladded uranium dioxide (UO2) fuel rod designs with reactor core burnups up to a maximum rod-average of ~~68~~80 GWd/MTU (and fuel enrichments up to 8 weight-percent uranium-235~~), including chromium-coated cladding (thicknesses less than 50 μm)~~ for PWRs and ~~chromia-doped (up to 0.16~~10 weight-percent for BWRs), for currently approved (as of issuance of this RG revision) fuel~~. SAND2011-0128 is not applicable for other fuel-clad combinations, including fuel with FeCrAl~~ and cladding materials, examples

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

14 In this guide, "table 1" refers collectively to tables 1.1, 1.2, and 1.3.

of which includes coated zirconium alloy ~~cladding~~claddings and doped fuels. Lastly, the NRC internal memorandum "Letter Report on Evaluation of the Impact of Fuel Fragmentation, Relocation, and Dispersal for the Radiological Design Basis Accidents in Regulatory Guide 1.183," dated July 20, 2021 (Ref. 27), assesses the impact of fuel fragmentation, relocation, and dispersal behavior on the accident source terms from SAND2011-0128. The conclusions of the July 20, 2021, memo also extends to SAND2023-01313 source term. Based on information provided in that memo, for the purposes of assessing the radiological consequences of the MHA LOCA, the impact of fuel fragmentation, relocation, and dispersal does not need to be considered for the range of applicability of burnups and enrichments in ~~SAND2011-0128. This RG does not provide guidance related to an acceptable treatment of fuel dispersal during non-LOCA DBAs.~~SAND2023-01313.

~~***Table 1. BWR Core Inventory Fraction Released into Containment Atmosphere***~~

Pathway-Specific Release Fractions using Mechanistic Transport Modeling[KM-05]

For the BWR AST, separate pathway-specific release fractions are provided in table 1 for the containment leakage pathway, the main steam line leakage pathway through the MSIVs, and for the liquidus leakage pathway from the suppression pool water volume. The purpose of breaking the BWR AST into three pathways is to consider the retention of fission products in the suppression pool in the determination of release fractions. The containment source term in Table 1.1 represents the cumulative containment inventory fraction excluding the suppression pool inventory in each phase of MHA LOCA. The suppression pool source term in Table 1.2 represents the cumulative suppression pool inventory fraction in each phase of the MHA LOCA. In contrast to these, the steam line source term in Table 1.3 represents the *time-averaged*, airborne core inventory fraction present in the steam line over each phase in the volume of the steam line that lies between the first SRV and the first MSIV over all steam lines. See Appendix A for additional information.

Table 1.1 BWR Core Inventory Fraction Released into Containment Atmosphere

Table 1.1

Group	Gap Release Phase	Early In-Vessel Phase	~~Total~~
Noble Gases	1.6x10-2	9.5x10-1	
Halogens	1.3x10-6	6.0x10-2	
Alkali Metals	1.2x10-6	6.0x10-3	
Tellurium Metals	<1x10-6	3.8x10-2	
Barium, Strontium	<1x10-6	3.0x10-4	
Noble Metals	<1x10-6	7.4x10-6	
Cerium Group	<1x10-6	<1x10-6	
Lanthanides	<1x10-6	<1x10-6	
Molybdenum	<1x10-6	1.0x10-4	

Deleted Cells

Table 1.2 BWR Core Inventory Fraction Retained in the Suppression Pool

Group	Gap Release Phase	Early In-Vessel Phase
Noble Gases	<1x10-6	<1x10-6
Halogens	5.0x10-3	6.5x10-1
Alkali Metals	5.0x10-3	3.1x10-1
Tellurium Metals	3.0x10-3	5.2x10-1
Barium, Strontium	6.0x10-4	4.7x10-3
Noble Metals	<1x10-6	6.0x10-3
Cerium Group	<1x10-6	<1x10-6
Lanthanides	<1x10-6	<1x10-6
Molybdenum	1.9x10-5	1.2x10-1

Table 1.3 BWR Core Inventory Time-Averaged Fraction Released into Steam Line[15]

[15] This represents the core inventory released to the portion of the steam line that lies between the first SRV and the MSIV. It is the time-averaged airborne fraction summed over all steam lines.

Table 1.3

Group	Gap Release Phase[16]	Early In-Vessel Phase[14]
Noble Gases	2.9x10-5	1.1x10-3
Halogens	5.6x10-6	5.1x10-5
Alkali Metals	5.1x10-6	1.3x10-5
Tellurium Metals	3.2x10-6	2.7x10-5
Barium, Strontium	6.1x10-7	2.4x10-7
Noble Metals	<1x10-9	2.4x10-7
Cerium Group	<1x10-9	<1x10-9
Lanthanides	<1x10-9	<1x10-9
Molybdenum	3.3x10-9	3.0x10-6

Table 2. PWR Core Inventory Fraction Released into Containment Atmosphere

Table 2

Group	Gap Release Phase	Early In-Vessel Phase	Total
Noble Gases	2.6x10-2	9.3x10-1	
Halogens	7.0x10-3	5.8x10-1	
Alkali Metals	3.0x10-3	5.0x10-1	
Tellurium Metals	6.0x10-3	5.5x10-1	
Barium, Strontium	1.0x10-3	2.0x10-3	
Noble Metals	<1x10-6	8.0x10-3	
Cerium Group	<1x10-6	<1x10-6	
Lanthanides	<1x10-6	<1x10-6	
Molybdenum	2.0x10-5	1.5x10-1	

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

The applicability of the steady-state fission product release fractions in tables 3 and 4 is limited to currently approved (as of the issuance of this RG) full-length UO_2 fuel rod designs operating up to a maximum rod-average burnup of ~~68~~80 GWd/MTU at power levels below the burnup-dependent power envelopes depicted in ~~figure~~figures 1 and 2. In ~~figure~~figures 1 and 2, the bounding rod-average power

[16] The release fractions in this table already take fission product deposition into account so credit should not be taken for deposition inboard of the first MSIV during the gap and early in-vessel phases.

refers to the rod-average linear heat generation rate of the peak rod, and the peak power refers to the maximum local linear heat generation rate in the core. Licensees should make adjustments to account for power uncertainties and plant maneuvering when comparing operating power histories to ~~figure~~figures 1 and 2. If it can be demonstrated that local power level, rate of fission gas release, and cumulative fission gas release remain less than those of the limiting co-resident UO_2 fuel rod, then the steady-state fission product release fractions in tables 3 and 4 apply to fuel rod designs containing integral burnable absorbers (e.g., gadolinia). One acceptable means of demonstrating this is by using an NRC-approved fuel performance code that has fission gas release models that are applicable to the integral burnable absorber fuel designs. If BWR part-length fuel rods are treated as full-length fuel rods with respect to overall quantity of fission products and are operated below the burnup-dependent BWR peak power envelope in figure 1, then table 3 steady-state fission product release fractions apply to these part-length fuel rod designs. Applicability to future fuel rod designs, including Cr-coated zirconium (Zr) cladding, non-Zr claddings, doped UO_2 fuel, high-density fuel, and mixed-oxide fuel, will be evaluated on a case-by-case basis. Appendix I provides an acceptable analytical technique for calculating plant-specific or fuel-rod-design-specific fission product release fractions.

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

Group	Fraction
I-131	0.~~03~~04
I-132	0.~~03~~04
Kr-85	0.~~32~~39
Other Noble Gases	0.~~03~~04
Other Halogens	0.~~02~~03
Alkali Metals	0.~~16~~20

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

	14 x 14	17 x 17
Group	**Fraction**	**Fraction**
I-131	0.~~07~~09	0.03
I-132	0.~~07~~10	0.03
Kr-85	0.~~40~~51	0.49
Other Noble Gases	0.~~06~~08	0.02
Other Halogens	0.~~04~~05	0.02
Alkali Metals	0.~~20~~26	0.25

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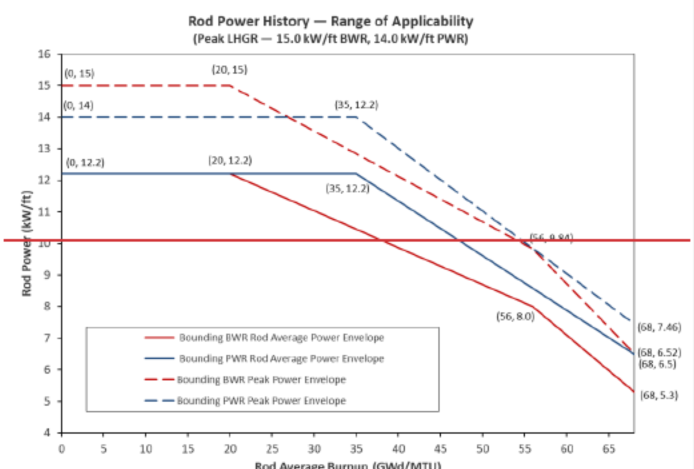
Rod Power History — Range of Applicability
(Peak LHGR — 15.0 kW/ft BWR, 14.0 kW/ft PWR)
Rod Power (kW/ft)
Rod Average Burnup (GWd/MTU)
(0, 15)
(20, 15)
(0, 14)
(35, 12.2)
(0, 12.2)
(20, 12.2)
(35, 12.2)
(56, 8.0)
(68, 7.46)
(68, 6.52)
(68, 6.5)
(68, 5.3)
Bounding BWR Rod Average Power Envelope
Bounding PWR Rod Average Power Envelope
Bounding BWR Peak Power Envelope
Bounding PWR Peak Power Envelope

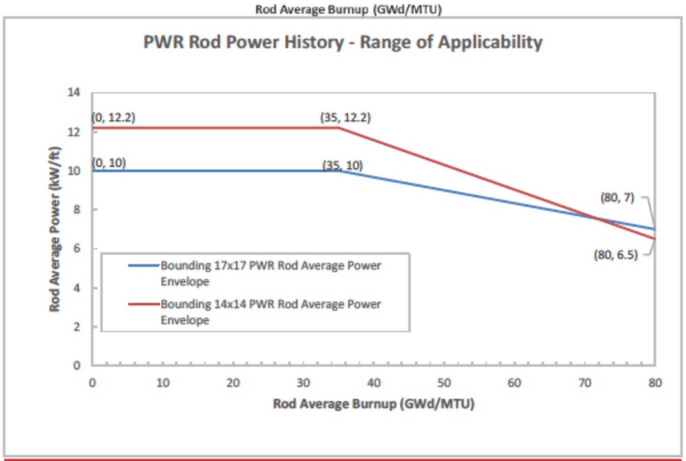
PWR Rod Power History - Range of Applicability
Rod Average Power (kW/ft)
Rod Average Burnup (GWd/MTU)
(0, 12.2)
(35, 12.2)
(0, 10)
(35, 10)
(80, 7)
(80, 6.5)
Bounding 17x17 PWR Rod Average Power Envelope
Bounding 14x14 PWR Rod Average Power Envelope

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

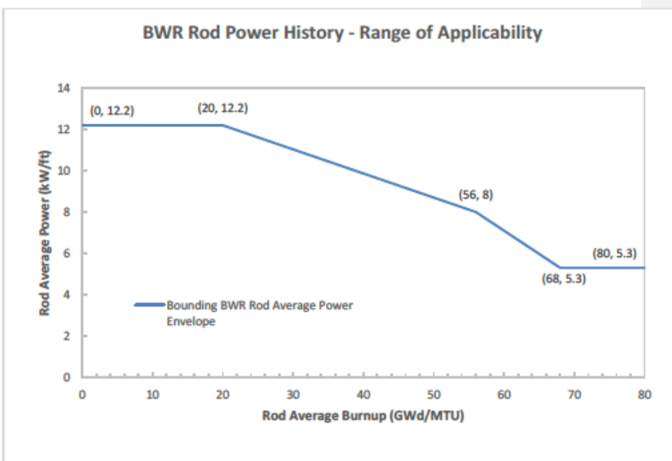

Figure 2. Maximum allowable power operating envelope for BWR steady-state release fractions

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

pellet burnup $<$ 50 GWd/MTU,

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

pellet burnup $\geq$ 50 GWd/MTU,

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

where

T_{FGR} = transient fission gas release fraction, and

ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

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

pellet burnup < 50 GWd/MTU,

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

pellet burnup $\geq$ 50 GWd/MTU,

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

where

T_{FGR} = transient fission gas release fraction, and

ΔH = increase in radial average fuel enthalpy, Δ calories per gram.

For the remaining high temperature non-LOCA DBAs that predict fuel rod cladding failure, such as the PWR reactor coolant pump locked rotor and ~~fuel handling accident~~main steam line break, additional fission product releases may occur as a result of fuel pellet fragmentation (e.g., fracturing of high-burnup rim region) due to loss of pellet-to-cladding mechanical constraint or impact loads. T_{FGR} has been experimentally observed under a variety of accident conditions. At the time of issuance of Revision ~~1~~2 of this RG, no consensus exists on the mechanism or the computation of T_{FGR} for these events; therefore, ~~future applicants should address this using engineering judgment or experimental data.~~an acceptable method to address T_{FGR} for non-LOCA DBAs other than RIA would be to prevent balloon and burst failures though design and analysis. Though not fully applicable to non-LOCA and non-reactivity-initiated DBAs, NRC Research Information Letter 2021-13, "Interpretation of Research on Fuel Fragmentation Relocation, and Dispersal at High Burnup," issued December 2021 (Ref. 29), provides data that can be used to provide a bounding estimate of T_{FGR} for high-temperature DBAs.

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

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

3.3 Timing of Release Phases

Table 5 provides the onset and end time of each sequential release phase for LOCA DBAs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel release phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase.[17] 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	End Time	Onset	End Time
Gap Release	0.5 minutes	~~0.23~~1.3 hours	2 minutes	0.~~19~~7 hours
Early In-Vessel	~~0.23~~1.3 hours	4.~~5~~0 hours	0.~~19~~7 hours	~~8.0~~6.7 hours

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

3.4 Radionuclide Composition

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

Table 6. Radionuclide Groups

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

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

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

Accident or Case	EAB and LPZ Dose Criteria (TEDE)	Control Room Dose Criteria[2] (TEDE)	Analysis Release Duration
MHA LOCA	0.25 sievert (Sv) (25 rem)	~~0.05 Sv (5.0 rem)~~See Table 8	30 days for containment, ECCS, and MSIV (BWR) leakage
BWR Main Steamline Break			Instantaneous puff
Fuel Damage or Pre-Accident Spike	0.25 Sv (25 rem)	0.~~05~~[[]] Sv (~~5~~[[]].0 rem)	
Equilibrium Iodine Activity	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
BWR Rod Drop Accident	0.063 Sv (6.3 rem)	0.~~05~~[[]] Sv (~~5~~[[]].0 rem)	24 hours
PWR Steam Generator Tube Rupture			Affected steam generator: time to ~~isolate~~[3]isolate
Fuel Damage or Pre-Accident Spike	0.25 Sv (25 rem)	0.~~05~~[[]] Sv (~~5~~[[]].0 rem)	Unaffected steam generator(s): until shutdown cooling is in operation and releases from the steam generator have been terminated
Concurrent Iodine Spike	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
PWR Main Steamline Break			Until shutdown cooling is in operation and releases from the steam generators have been terminated
Fuel Damage or Pre-Accident Spike	0.25 Sv (25 rem)	0.~~05~~[[]] Sv (~~5~~[[]].0 rem)	
Concurrent Iodine Spike	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	
PWR Locked Rotor Accident	0.025 Sv (2.5 rem)	0.05 Sv (5.0 rem)	Until shutdown cooling is in operation and releases from the steam generators have been terminated
PWR Control Rod Ejection Accident	0.063 Sv (6.3 rem)	0.~~05~~[[]] Sv (~~5~~[[]].0 rem)	Containment pathway: 30 days; Secondary system: until shutdown cooling is in operation and releases from the steam generators have been terminated
Fuel Handling Accident	0.063 Sv (6.3 rem)	0.05 Sv (5.0 rem)	30 days

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

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

~~[a]~~[3] A graded, risk-informed and performance-based framework has been established for the control room dose criteria. The framework is applicable to the MHA-LOCA accident.

The licensee may assume credit for ESFs that mitigate airborne radioactive material within the control room. Such features may include control room isolation or pressurization or intake or recirculation filtration. Guidance appears in SRP Section 6.5.1, "ESF Atmosphere Cleanup Systems," and RG 1.52, Revision 4, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," issued September 2012 (Ref. 35). The control room design is often optimized for the MHA LOCA, and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by ESF signals or radiation monitors. In some cases, the ESF signal is effective only for selected accidents, placing reliance on the radiation monitors for the remaining accidents. Several aspects of radiation monitors can delay control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

4.2.5 Personal Protective Equipment

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

4.2.6 Dose Receptor Occupancy Factor

~~The~~Occupancy factors are used to estimate the amount of time a receptor, or individual, is present at a particular location, to estimate the amount of time a receptor was exposed to a radioactive source (e.g., plume). An occupancy factors is typically expressed in terms of fraction per day. With detailed information of a receptor's habits an analyst can determine appropriate locations and occupancy factors required for the analysis.[KM-07]

Absent facility-specific receptor information on occupational habits within the control room during an event, the dose receptor for these analyses is the hypothetical maximum exposed individual ~~who~~. The default occupancy factors assume the receptor 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.[21] For the duration of the event, the licensee should assume the breathing rate of this individual to be 3.5×10^{-4} m^3/s (Ref. 38).

Licensees develop and maintain staffing plans to execute their emergency plans. These plans can consider facility-specific considerations that impact the licensee's ability to staff emergency planning functions. Licensees can credit facility-specific staffing plans to define facility-specific occupancy factors that differ from the default values provided earlier in this section. The provision acknowledges the unique requirements of each facility and fosters alignment with other regulatory requirements that apply during accident scenarios. Facility-specific occupancy factors should align with the licensing basis and appropriately account for uncertainties (e.g., shift changes) with the intent of maintaining operational flexibility during an event.

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,

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

4.4 Acceptance Criteria

The accident dose radiological criteria for the EAB, for the outer boundary of the LPZ, and for the control room are in 10 CFR 50.34, 10 CFR Part 52, 10 CFR 50.67, and GDC 19 in Appendix A to 10 CFR Part 50. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation. For events with a higher probability of occurrence ~~(e.g., fuel handling accidents),~~, postulated EAB and LPZ doses should not exceed the criteria in table 7.1. 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. To support the increased enrichment rulemaking efforts, a graded, risk-informed and performance-based framework has been developed for the control room for the MHA-LOCA.[KM-08] This framework is presented in table 8. (see Ref. XX for a background on this framework methodology) The framework leverages facility specific risk-insights based on a pedigreed probabilistic risk assessment model.

In accordance with 10 CFR 50.67 and GDC-19, licensees may use a 10-rem TEDE control room design criteria value without further justification. The licensee may establish a control room design criterion value higher than [[]] rem TEDE but not greater than 25 rem TEDE for compliance provided the licensee demonstrates that the specified value is commensurate with the risk profile of the plant. The staff deem acceptable assessments that leverage risk-insights from probabilistic risk assessments for which the models are of the highest pedigree. Acceptability of the model is determined with respect to the following aspects: scope, level of detail, conformance to consensus standard technical elements (i.e., technical robustness), and plant representation. These models would be consistent with the philosophy in Regulatory Guide 1.174 and technical adequacy expectations for the model in Regulatory Guide 1.200. For instance, the use of all-hazard core damage frequency results from approved Technical Specifications Task Force (TSTF) Traveler TSTF-505, "Provide Risk-Informed Extended Completion Times – RITSTF Initiative 4b," license amendment request would be acceptable. The all-hazard core damage frequency is computed by summing all core damage frequency results (e.g., internal, flood, fires, seismic, high winds, and others).

For licensees with a high pedigree probabilistic risk assessment model, follow each step:

1. Select the corresponding facility specific control room design criteria in Table 8 based on the all-hazard core damage frequency.
2. Present a summary of the probabilistic risk assessment model (e.g., original submittal and staff safety evaluation) and any adjustments made to the model since the initial approval.
3. Reassess the facility-specific control room design criteria, as necessary, for subsequent license amendment request which impact the MHA-LOCA.

Table 8 Guidelines for Control Room Location Graded, Risk-Informed and Performance Based Framework

All-hazard CDF	Graded, Control Room Design Criteria (rem TEDE)
CDF < 1.E-5	[[]]
1E-5 < CDF < 5E-5	[[]]
5E-5 < CDF < 1E-4	[[]]
CDF > 1E-4; or, licensee not adopting the graded framework to determine acceptance criteria	[[]]

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

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

5. Analysis Assumptions and Methodology

5.1 General Considerations

5.1.1 Analysis Quality

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

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

5.1.2 Credit for Engineered Safeguard Features

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

5.1.3 Assignment of Numerical Input Values

The licensee should select the numerical values to be used as inputs to the dose analyses with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but nonconservative in another portion of the same analysis. For example, an assumption of minimum containment system spray flow is usually conservative for estimating iodine scrubbing but, in many cases, may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a given parameter for the duration of the event, particularly for parameters affected by changes in density. For a parameter addressed by technical specifications, the value used in the analysis should be that identified in the technical specifications.[22] 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.

Best-estimate plus uncertainty methods provide more realistic representations of uncertainty when compared to point estimate deterministic methods. This additional realism in the representation of uncertainty can enhance the accuracy of analyses. Methods for quantification of uncertainty in engineering models are well established. For example, one common method for uncertain analysis uses Monte Carlo statistical techniques which incorporate a random sampling of representative distributions of various model parameter inputs. Important considerations with respect to the application of uncertainty methods include the selection of parameters that are appropriate to model and distributions to be used in the analysis. When using best-estimate plus uncertainty techniques in the context of this guide, the proposed distributions may be based on measurements or on justifiable engineering judgement when data are not available. Descriptive statistics from the final distribution results should be provided with submittals applying these techniques and values from the 95-percentile should be used when demonstrating compliance. Sampling from within the parameter distributions must encompass the entire range of values to ensure a comprehensive representation of potential values. Extrapolation from the data will not be accepted, nor will triangular distributions. When performing such an analysis, a sufficient number of iterations must be performed to ensure a high level of confidence in the accuracy and reliability of the obtained results. A 95/95 confidence level is considered acceptable for comparing best-estimate predictions where the uncertainty analysis demonstrates with 95% confidence that 95% of the

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

results will be below the applicable acceptance criteria. Best-estimate plus uncertainty approaches that apply statistical sampling techniques, and parameter distributions, will be evaluated on a case-by-case basis. It is expected that a best-estimate plus uncertainty approach would produce results with a considerable margin to the regulatory acceptance criteria.

5.1.4 Applicability of Prior and the Proposed Licensing Basis

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

5.2 Accident-Specific Assumptions

The appendices to this RG provide accident-specific assumptions that are acceptable to the staff for performing site-specific analyses as required by 10 CFR 50.34, 10 CFR Part 52, 10 CFR 50.67, and GDC 19. Licensees should review their licensing basis documents for guidance on the analysis of radiological DBAs other than those provided in this guide. The DBAs addressed in these attachments were selected from accidents that may involve damage to irradiated fuel. This guide does not address all DBAs with radiological consequences. The inclusion or exclusion of a particular DBA in this guide does not mean that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST and changes to the facility or to the radiological analyses.

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

5.3 Atmospheric Dispersion Modeling and Meteorology Assumptions

Atmospheric dispersion factors (χ/Q values) for the EAB, the LPZ, the control room, and, as applicable, the onsite emergency response facility (i.e., the TSC)[23] that the staff approved during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified in this guide, provided that such values remain relevant to the particular accident, release characteristics that affect plume rise, its release points, and receptor locations. Licensees should ensure that any previously approved values remain accurate and do not include any misapplication of a

23 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.

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. 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." The NRC's regulatory framework ensures that licensees design emergency core cooling systems (ECCS) to meet acceptance criteria that limit peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation and that reactor cores remain in coolable geometries such that long-term cooling can be provided following a LOCA. Therefore, ECCS function to limit reactor core damage during a LOCA such that resultant radiological consequences are very low, especially when compared to acceptance criteria that apply to the MHA LOCA.

The MHA LOCA, like all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge aspects of the facility design. Specifically, the MHA LOCA is the DBA that is used to evaluate the siting of new reactors and the design of systems, structures and components (SSCs) that function to limit fission product releases to the environment (e.g., containments). Since light water reactors, with their ECCS, are designed such that LOCAs are not expected to result in significant radiological releases to the environment, the NRC regulatory framework requires that MHA LOCA analyses assume a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products. This approach ensures that the designs of SSCs that function to limit fission product releases are appropriately analyzed. .

The MHA LOCA~~Separate mechanistic analyses are performed using a spectrum of break sizes to evaluate fuel and emergency core cooling system performance for conformance with 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." With regard to radiological consequences, an MHA LOCA is typically assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility. The~~ analysis should calculate the limiting dose consequences to the public and control room doses, assuming a deterministic substantial core damage source term, discussed below, released into an intact containment.

A-1. Source Term

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

(particularly in BWR Mark I and Mark II drywells) should be considered in the determination of decontamination factors (DFs) and removal coefficients credited for the drywell or containment. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed region volume per hour, unless other rates are justified. On a case-by-case basis, the licensee may consider containment mixing rates determined by the cooldown rate in the sprayed region and the buoyancy-driven flow that results. The containment building atmosphere may be considered a single well-mixed volume if the spray covers at least 90 percent of the containment building space and an engineered-safety-feature (ESF) ventilation system is available for adequate mixing of the unsprayed compartments.

As provided in Section 6.5.2 of the SRP, the maximum DF for elemental iodine is based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached~~.~~ and that the elemental iodine removal is limited to 20 hr^{-1}. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in tables 1 and 2 of this guide, multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., the SRP methodology treats aerosols as particulate).

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

A-2.5 Historically, reduction in airborne radioactivity in the containment by suppression pool scrubbing has not been credited in licensing actions for operating BWRs; however, ~~the staff may consider such reduction on a case-by-case basis.[1] The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. A-8).~~ Section 3.2 now provides release fractions for pathway-specific release fractions using mechanistic transport modeling that the staff consider acceptable. For suppression pool solutions having a pH less than 7, elemental iodine vapor should be conservatively assumed to evolve into the containment atmosphere.

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

A-2.7 The evaluation should assume that the primary containment (~~i.e., drywell and~~including the 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

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

the first 24 hours to 50 percent of the TS leak rate. The leak rate may be reduced after the first 24 hours to 50 percent of the TS leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50 percent of the TS leak rate. Leakage from sub-atmospheric containments is assumed to terminate when the containment is brought to and maintained at a sub-atmospheric condition as defined by the TS.

The licensee's evaluation of the post-accident containment pressure response may credit safety related systems to further reduce containment pressure, and thereby the associated containment leak rate, in the first 24 hours, and for the duration of the accident. Section 5.1.2 addresses credit for engineered safeguard features.

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

A-3. Dual Containments

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

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

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

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

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

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

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

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

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

A-4.1 With the exception of noble gases, all fission products released from the fuel to the containment (as defined in the applicable tables 1 ~~and~~through 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~~For BWRs, Table 1.2 provides a mechanistic model of the core inventory fraction retained in the suppression pool based on analyses from SAND2023-01313. Table 1.2 is a suitably conservative mechanistic model for the transport of airborne activities in the suppression pool. For PWRs, in lieu of the deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameter values that make spray and deposition models conservative in estimating containment airborne leakage are nonconservative in estimating the buildup of sump activity.

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

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

return to the refueling water storage tank in the emergency core cooling system) should also be considered.

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

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

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

where

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

A-4.5 If the temperature of the leakage is less than 212°F or the calculated FF is less than 10 percent, the amount of iodine that becomes airborne should be assumed to be 10 percent of the total iodine activity in the leaked fluid, unless a smaller amount can be substantiated. The justification of such values should consider the sump pH history; changes to the leakage pH caused by pooling on concrete surfaces, leaching through piping insulation, evaporation to dryness, and mixing with other liquids in drainage sumps; area ventilation rates and temperatures; and subsequent re-evolution of iodine.

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

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

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

~~Three~~Two methods are presented below to compute aerosol deposition within main steamlines. Each method computes similar removal coefficients that are suitable for radiological consequence

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

calculations. These methods are not valid if credit is also taken for aerosol removal from drywell sprays, or for other containment aerosol removal processes, when modeling the MSIV leakage release pathway without accounting for the change in particle size distribution due to these containment removal processes. The ~~three~~two MSIV leakage models are the following:

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

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

~~c.~~b. numerical integration.

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

~~A-5.1 The source of the MSIV leakage should be assumed to be the containment (or drywell)[4] activity concentration (see Regulatory Position A-2.1).~~A-5.1 The source of the MSIV leakage should be assumed to be the BWR core inventory time-averaged fraction released into steam line presented in Table 1.3. This represents the time-averaged overall fraction of the fission product inventory in the portion of the steamline just downstream of the first SRV to the first MSIV in all steam lines. To ascertain the fission product concentration, C, for the region-specific BWR source term analyses, the licensee should divide by their plant-specific volume of this portion of the steam line multiplied by the number of steam lines.

Table 1.3 is a suitably conservative mechanistic model for the transport of airborne radionuclides in the steam lines. When using this approach, no credit should be taken for aerosol deposition upstream of the in-board MSIV before the end of the early in vessel release phase. Up to the end of the early in vessel phase (i.e., the gap and early in-vessel phases), the concentration, C, of aerosols should be modeled as a constant source. After the early in-vessel release phase, credit for aerosol deposition may be credited as aerosols and steam condense from contact with cooling surfaces. Modeling aerosol deposition may take on the form as an exponentially declining input rate as, $C(t) = C(0)e^{-\lambda t}$, where λ is the rate constant having units of inverse time (t^{-1}). Based on the region-specific analysis and subsequent assessments (Ref. XX), an acceptable rate constant should be modeled as 1.2E-1 hr^{-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 developing the guidance for BWR Mark I, II, and III plants may be followed as applicable to designs under consideration.

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

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

of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.

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

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

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

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

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

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

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

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

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

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

- a discussion of seismic capacity and margin present in the relevant SSCs, including the condenser, based on their design code(s) of record,
- insights from the Individual Plant Examination for External Events as described above,
- walkdown(s) of the SSCs in the alternate pathway, including the condenser, performed by knowledgeable licensee staff members, to ensure that items adversely affecting the seismic capacity of relevant SSCs (e.g., loose or missing anchorages and degraded pipe supports) are identified and corrected, and
- confirmatory calculations for a sample of piping supports, to verify that they provide acceptable flexibility at terminal ends of piping and major branch connections.

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

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

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

5 The credit described in this regulatory position will supersede the aerosol settling estimates previously given in the NRC staff document AEB 98-03 (Ref. A-13) when ~~Revision~~Revisions 1 and 2 of RG 1.183 is used.

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

~~Time~~	~~Inboard~~	~~Between MSIVs~~	~~Outboard~~	~~Condenser~~
~~(hr)~~	~~(hr⁻¹)~~	~~(hr⁻¹)~~	~~(hr⁻¹)~~	~~(hr⁻¹)~~
~~0–10~~	~~0.0~~	~~1.8~~	~~1.0~~	~~0.015~~
~~10+~~	~~0.0~~	~~1.0~~	~~0.7~~	~~0.012~~

~~A-5.6.2~~ Reevaluated AEB-98-03 with the multi-group method: ~~Aerosol~~At the beginning of the accident, aerosol deposition removal coefficients for the main steamline piping between the MSIVs and downstream of the MSIVs may apply an updated AEB 98-03 use of the Stokes settling velocity physics parameters with the multi-group method. After the early in-vessel release phase, credit for aerosol deposition upstream of the in-board MSIV may also be credited as steam condenses from contact with cooling surfaces as described in Regulatory Position A-5.1. The method below computes both total effective aerosol removal efficiencies (TEAREs) (i.e., filter efficiencies) and equivalent removal coefficients λ (hr^{-1}).

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

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

where

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

The document "State-of-the-Art Report on Nuclear Aerosols," issued in 2009 (Ref. A-19), provides a summary of experimental observations from integral experiments involving irradiated fuel to infer characteristics of aerosols under light-water reactor severe accident conditions. The State-of-the-Art Report recommends the use of a log-normal distribution for aerosols in the RCS (AMMD 1.0 microns (μm) with a geometric standard deviation, σ_g, of 2.0), and provides PHÉBUS-Fission Product aerosol measurements in containment (AMMD of 3.0 μm and σ_g of 2.0). Considering the MHA LOCA modeling approach, which considers no pipe break and where the deposition properties after reflood are based on the characteristics of the RCS and containment aerosol (i.e., the approach considers the effects of an active emergency core cooling system), the methods in Regulatory Positions A-5.6.~~2~~1 and A-5.6.~~3~~2 should assume a log-normal aerosol diameter distribution with an AMMD of 2.0 μm and σ_g of 2.0. Assume as fixed values a $C_s(d_a)$ of 1 and a viscosity of 1.93x10^{-5} Pascal-second. At least 10,000 trials are necessary to develop a settling velocity distribution dataset. Note that while the NRC memorandum of Reference A-20 addresses the methods discussed in Regulatory Positions A-5.6.2 and A-5.6.3, it does not establish regulatory positions. For example, regarding Reference A-20, this memorandum does not endorse input parameters such as the AMMD, assumed in example calculations, and statements on the validity of the existing 20-group method.

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

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

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

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

where

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

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

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

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

where

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

A-5.6.~~3~~2 Numerical Integration: ~~Aerosol~~At the beginning of the accident, aerosol deposition removal coefficients for the main steamline piping between the MSIVs and downstream of the MSIVs may apply the method's use of the Stokes settling velocity and physics parameters of Regulatory Position A-5.6.~~2~~1 with the numerical integration method. After the early in-vessel release phase, credit for aerosol deposition upstream of the in-board MSIV may also be credited as steam condenses from contact with cooling surfaces as described in Regulatory Position A-5.1. The equation for a normalized number distribution ($n(d_a)$) of particles of aerodynamic diameter (d_a) is given (Ref. A-21) as follows:

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

where

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

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

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

Discretize the range of particle diameters, d_a, from 1x10^{-9} µm to 1x10^{-3} µm into 150 groups. For each group, apply equation A-4 to compute the normalized number distribution, ($n(d_a)$). Then, for each discretized group, (1) compute its settling velocity by applying equation A-1, (2) use equation 3 from Reference A-20 to compute inboard and outboard concentrations of particles leaving the volumes, (3) sum up the inboard and outboard concentrations using an appropriate numerical integration technique (such as the trapezoidal method), and (4) use equation A-2 to then compute the filter efficiencies. Finally, use equation A-3 to convert filter efficiencies, η_{filt}, into removal coefficients (in units of hr^{-1}).

A-5.6.~~4~~3 Aerosol deposition removal coefficients for the condenser using a multi-group method and numerical integration are acceptable and will be evaluated on a case-by-case basis.

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

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

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

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

A-6. Containment Purging

The licensee should analyze the radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure. If the installed containment purging

APPENDIX I

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

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

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

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

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

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

Note: This document contains draft/predecisional guidance language to support discussions during the NRC's public meeting on June 12, 2024 (ML24155A115).The NRC staff are not officially accepting comments during the meeting and will not provide any formal responses to any feedback provided during this meeting

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

I-1.1

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

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

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

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

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

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

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

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

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

where

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

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

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

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

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

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

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

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

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

I-3. ~~High-confidence upper tolerance release~~ Release fractions should be calculated using an NRC-approved fuel rod thermal-mechanical code, ~~along with quantified model uncertainties.~~ which has a NRC-approved methodology for calculating high-confidence stable fission gas release and fuel temperatures.

Note: This document contains draft/predecisional guidance language to support discussions during the NRC's public meeting on June 12, 2024 (ML24155A115).The NRC staff are not officially accepting comments during the meeting and will not provide any formal responses to any feedback provided during this meeting

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

~~I-3.2~~I-3.1 When using an NRC-approved fuel rod thermal-mechanical code which may produce different values for R/B of Kr85m, an alternative factor may be derived following the same procedure as described in Section 4.4.1 and 4.4.2 of NUREG/CR-7003 with nominal predicted fuel rod temperatures. For long-lived isotopes, established model uncertainties associated with the NRC-approved fuel rod thermal-mechanical code should be applied, either deterministically or sampled within a statistical application methodology, to obtain high-confidence upper tolerance release fractions.

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

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

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

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

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

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