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Comment On: NRC-2021-0179-0001
Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

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General Comment

See attached file(s)

Attachments

06-21-22_NRC_Industry Comments on DG-1389

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Subject: NEI Comments on draft Regulatory Guide (DG), DG-1389, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Docket ID NRC-2021-0179.

Submitted via regulations.gov

Dear Program Management, Announcements and Editing Staff,

The Nuclear Energy Institute (NEI)[1], on behalf of our members, appreciates the opportunity to provide comments on the subject draft regulatory guide (DG), DG-1389, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." This DG is proposed Revision 1 to Regulatory Guide (RG) 1.183 which describes a method that the NRC staff considers acceptable for complying with regulations for design basis accident dose consequence analysis using an Alternative Source Term. The purpose of this letter is to provide the attached comments which recommend several changes to improve clarity and consistency on the recommended approaches, methods, and analysis this guidance provides. These comments were developed by a nuclear energy industry task force and reflect a substantial body of industry technical expertise, experience, and lessons learned gained from successful licensing actions utilizing alternative radiological source terms as part of their supporting design basis accident analyses.

Of particular concern is that the DG does not include the results of the staff's PWR and BWR analyses to enable an assessment of the impact to analyzed doses resulting from the updated guidance, such as impact from the new release fractions and timing. Industry views the proposed changes in this DG just as significant as when the guidance transitioned from the TID Source Term to the Alternative Source Term. As proposed, the conservatisms and changes incorporated in this revision precludes its use by many plants that

[1] The Nuclear Energy Institute (NEI) is responsible for establishing unified policy on behalf of its members relating to matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect and engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations involved in the nuclear energy industry.

are interested in implementing near-term ATF design concepts, fuel burnup extension to 68 GWd/MTU (peak rod average), and 235U enrichments up to 8.0 wt%.

A comparison of the changes in MHA release fractions between BWRs (Table 1) and PWRs (Table 2) identified a significant increase in the BWR halogen release fractions with no indication in either SAND2011-0128 or DG-1389 as to the cause. This increase may adversely affect the ability of BWRs to comply with RG 1.183 Rev. 1. A comparison to the PWR analyses suggests that the accident sequences may be responsible for this impact. SAND2011-0128 updates NUREG-1465 with higher core exposures utilizing the latest NRC's MELCOR methodology. The accident sequences that were analyzed to develop the PWR release fractions are listed in Table 5 of the Sandia report and include a variety of accident types. However, for the BWR release fractions, Table 3 of the Sandia report indicates that nearly all the evaluations were based on station blackout (SBO) sequences. However, the risk from SBO events has been substantially reduced by the industry's implementation of FLEX. Consistent with the PWR analysis in SAND2011-0128, the BWR release fractions should be re-evaluated to ensure that they are based on an appropriate set of accident sequences that more accurately reflect BWR risk profiles.

Additionally, it is unclear from the SAND2011-0128 report whether suppression pool scrubbing was credited in determining the release fractions from a BWR SBO. Credit for suppression pool scrubbing can significantly decrease the airborne activity since the SBO-related releases would be released via spargers submerged in the suppression pool. Therefore, any sequences that involve releases through the pool spargers should take credit for suppression pool scrubbing.

Also, although new guidance was added for crediting holdup and retention of MSIV leakage within the main steamlines and condenser for BWRs, it contains more conservative assumptions than previous models that limits its effectiveness. For example, Section A-5 of the DG presents three acceptable methods for calculating aerosol deposition within the main steam lines, but also states, "…these methods are not valid if credit has been taken for aerosol removal from drywell sprays." No technical justification is provided for why aerosol removal from drywell sprays is not valid if used with credit for main steamline deposition. As a result, the effectiveness of the application of these models is significantly reduced. Considering the number of BWRs currently modeling both removal mechanisms, the DG should provide guidance for crediting both of these important mitigative features.

Further, the DG states, "Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations. A combination of the methods contained in revision 0 or revision 1 of RG 1.183 would need additional justification." However, it is not clear in the proposed document if licensees can transition one (or a few) analyses to revision 1, or if full implementation for all design basis accidents is required. Clarification should be added to discuss that selective implementation is acceptable, provided that each accident analysis uses either revision 0 or revision 1 and to specify that a combination of the methods contained in revision 0 or revision 1 of RG 1.183, in a single analysis, would need additional justification.

Finally, we appreciate the staff's efforts to include guidance for accident tolerant fuel (ATF), high-burnup fuel, and increased enrichment source term analyses and in revising the transport and decontamination models for the fuel handling DBA when developing this draft guidance. We encourage your consideration of all stakeholder comments prior to finalizing this draft RG. Given the long-lasting impact of the final document, we recommend a public meeting be scheduled to discuss in more detail the comments included in the attachment and how the staff plans to address them. We trust that you will find these comments useful and informative as you finalize the draft, and we look forward to future engagement on this important matter. Please contact me at fap@nei.org or (202) 739-8132 with any questions or comments about the content of this letter or the attached comments.

Sincerely,

Frances Pimentel

Frances A. Pimentel

Attachment

c: Sean Meighan, NRR/DRA/ARCB, NRC
Mike Franovich, NRR/DRA, NRC

Consolidated Comments on Draft Regulatory Guide DG-1389 (RG 1.183, Rev.1)

"Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

No.	Section	Comment/Basis	Recommendation
1.	Section B (Background) [p. 5] and Section 1.4 Risk Implications [p. 14-15]	The NRC has identified that licensees may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. The NRC should provide some more explanation on the parameters/limitations of using risk insights.	NRC should provide some additional explanation on the acceptable uses of risk insight in conjunction with the radiological analyses performed with this RG.
2.	Section B. (Background) [p. 6]	The last paragraph of the section states: "Revision 0 of RG 1.183 will continue to be available for use by licensees and applicants as a method acceptable to the NRC staff for demonstrating compliance with the regulations. A combination of the methods contained in Revision 0 or Revision 1 of RG 1.183 would need additional justification." The NRC should provide additional clarifications in this paragraph (or Section 1.2) to inform the licensee as to acceptable instances for combination of Revision 0 and 1 methods.	NRC should include a statement like the following: "For example, across a licensing basis, different revision methods may be adopted provided the individual analyses fully adopt a single revision (i.e., individual analysis inputs, assumptions, or modeling aspects may not adopt different revision methods within the same analysis)." Additional details should also be included in Section 1.2 – Scope of Implementation.
3.	Section 1.1.1 (Safety Margins) [p.8]	The last sentence states: "Once the staff has approved the initial AST implementation and it has become part of the facility design basis…" This sentence should be clarified to identify the initial AST implementation associated with Revision 1 of RG 1.183.	The NRC should clarify the sentence as: "Once the staff has approved the initial AST implementation in accordance with Revision 1 and it has become part of the facility design basis…"
4.	Section 1.3.2 (Reanalysis Guidance) [p.13]	SECY-98-154, Reference 19, performed a re-baselining of sample radiological consequences analyses when transitioning from TID to AST source terms. Has the NRC performed similar re-baselining studies for the major method changes being made in DG-1389 compared to RG 1.183, Revision 0? Notably, the NRC should investigate sample PWR and BWR analyses for the LOCA release fraction and timings, non-LOCA release fractions (including TFGR components), updated FHA modeling, etc. If this has already been performed, the wording for this section should be supplemented to include reference to these studies.	Supplement Section 1.3.2 by incorporating results from supporting analyses performed by staff when developing the proposed revisions contained in DG-1389.

No.	Section	Comment/Basis	Recommendation
5.	Section 3.1 (Fission Product Inventory) [p.17]	RG 1.183, Revision 0 identified specific computer codes acceptable for use in core inventory calculations (ORIGEN2 and ORIGEN-ARP). These code examples were deleted for DG-1389. The NRC should continue to include examples of core inventory codes which are acceptable for use.	The NRC should continue to include examples of core inventory codes which are acceptable for use in core inventory calculations.
6.	Section 3.1 Fission Product Inventory – [p.18]	Based on the directive to "maximize fission product inventory" it is implied that bounding core parameters should be used, however downstream discussion (e.g., Section 3.2) states that the inventory should be for an equilibrium core. The RG should be consistent in prescribing if a bounding or an equilibrium core should be applied to dose calculations.	The RG should be consistent in prescribing if a bounding or an equilibrium core should be applied to dose calculations.
7.	Section 3.1 Fission Product Inventory – [p.18]	It's not clear that this directive can be applied. Many BWR units do not have radial peaking factors in the COLR nor Technical Specifications.	Remove discussion of COLR/TS from this section.
8.	Section 3.2 (Release Fractions) [p.18]	Footnote 10 identifies that the data in this section does not apply to cores containing mixed oxide fuel (MOX). The section also identifies that Accident Tolerant Fuel (ATF) concepts, excluding near-term ATF concepts, are also not applicable to the data in this section. Footnote 10 should also include these ATF concepts as not being applicable for the data in Section 3.2.	Footnote 10 should also include these ATF concepts as not being applicable for the data in Section 3.2.
9.	Section 3.2, 4th paragraph [p.18]	It states the steady-state fission product release fractions in Table 3 can only be used if BWR part-length rods are treated as full-length rods with respect to overall quantity of fission products. BWR fuel bundles can have up to 20 part-length rods. Assuming all the part length rods have the same fission products of a full-length rod penalizes the source term by a large amount. The consequence of this requirement would be the source term has many pins more worth of inventory than it actually has.	Please remove the requirement that BWR part-length rods are treated as full-length rods with respect to overall quantity of fission products.
10.	Section 3.2 (Release Fractions) [p.18]	Regulatory Position 3.2 includes the following statement: "If it can be demonstrated that local power level, rate of fission gas release, and cumulative fission gas release remain less than the limiting co-resident UO2 fuel rod, then Table 3 and 4 steady-state fission product release fractions apply to fuel rod designs containing integral burnable absorbers (e.g., Gadolinia)."	The NRC should clarify the level of justification needed to confirm fission gas release rates and cumulative fission gas release for fuel designs containing integral burnable absorbers.
11.	Section 3.2 (Release Fractions) [p.19]	A factor is presented to adjust the transient fission product release correlations for short-lived isotopes. This factor is not prominently presented with the correlations and may be missed by applicants or licensees.	The NRC should present the factor for short-lived isotopes more prominently with the correlations presented in the section.

No.	Section	Comment/Basis	Recommendation
12.	Section 3.2 (Release Fractions) [p.19]	Section 3.2 contains the following guidance: *For the remaining non-LOCA DBAs which predict fuel rod cladding failure, such as PWR reactor coolant pump locked rotor and fuel handling accident, additional fission product releases may occur as a result of fuel pellet fragmentation (e.g., fracturing of high burnup rim region) due to loss of pellet-to-cladding mechanical constraint or impact loads. TFGR has been experimentally observed under a variety of accident conditions and should be addressed in future applications.* However, the regulatory guide does not provide guidance regarding an acceptable treatment for any additional fission product releases for these accidents. Therefore, the current guidance within the draft regulatory guide is incomplete.	Recommend that the current draft regulatory guide be updated to provide complete guidance for these remaining accidents.
13.	Section 3.2 (Release Fractions) [p.19]	The paragraph starting with "For the remaining non-LOCA DBs which predict fuel rod cladding failure…" is not clear on the background of the issue or the success pathways afforded to the licensees to address the NRC's concerns on the topic in question. Ideally, background documents for further description of the issue and/or documents identifying acceptable methods of evaluation are cited. The NRC should include these documents in order for the licensees to fully understand and properly address the issue in question. If no documents exist, the NRC should develop these positions.	Include background documents for further description of the issue and/or documents identifying acceptable methods of evaluation so that licensees can fully understand and properly address the issue in question. If no documents exist, the NRC should develop these positions.

No.	Section	Comment/Basis	Recommendation
14.	Section 3.2 (Release Fractions) [p.20]	Tables 1 and 2 present core inventory fraction releases into containment for BWRs and PWRs. The release groups are updated from Revision 0 of RG 1.183. One significant difference is that the tellurium group (both PWRs and BWRs) and barium, strontium group (PWRs only) identify a release fraction during the gap release phase. From page 34 of the SAND-2011-0128 report: "At the same time, the Gap Release Phase was calculated to be long enough that some core degradation characteristic of the In-vessel Phase of release as prescribed in the NUREG-1465 Source Term did take place. This is indicated by small amounts of tellurium release during the Gap Release Phase and in the case of PWR accidents small amounts of alkaline earth release. Ordinarily tellurium and alkaline earths are not thought to be contributors to the gap release. They contribute here because some portions of the core had entered into what would be categorized phenomenologically as in-vessel release before the criterion to terminate gap release had been reached." This confirms that, while the timing of the Te, Ba, and Sr groups does release during the gap fraction phase, these releases are more accurately associated with the early in-vessel phase. The continued identification of these groups' release as gap phase release creates confusion with subsequent gap fraction tables (Tables 3 and 4) which do not identify these nuclide groups as being released. **[Note that it is <u>not</u> requested to add Te, Ba, and Sr to the gap releases.]**	The NRC should clarify that the Te, Ba, and Sr groups do contribute releases during the gap release phase for LOCA releases in Tables 1 and 2; however, these releases are more accurately classified as early in-vessel releases and these groups do not need to be considered in gap releases for other events.
15.	Section 3.2 Tables 1 & 2 (p20), and Table 6 (p22)	Tables 1 and 2 report release fractions from a new Molybdenum group; however, the elements in this new group are not listed in Table 6. Which release group is Zirconium (Zr) considered to reside? DG-1389 is consistent with Revision 0 and indicates it is part of the Lanthanides while Table 14 in the underlying Sandia report (SAND 2011-0128) reports Zirconium as part of the Cerium group.	It is expected that the final grouping is consistent with SAND-2011-0128 since that is the basis for the release fractions in Tables 1 & 2. Update the Table 6 for the new proposed Molybdenum group. Change Zr grouping to Cerium group.

No.	Section	Comment/Basis	Recommendation
16.	Section 3.2 (Release Fractions) [p.21]	Figure 1 presents the maximum allowable power operating envelope for steady-state release fractions. The envelope is identified by rod average power (kW/ft) and rod average burnup (GWD/MTU). The figure further identifies a "Peak LHGR" (Linear Heat Generation Rate) of 15.0 kW/ft for BWRs and 14.0 kW/ft for PWRs. From the Ref. 24 Technical Basis for Non-LOCA Fission Product Release Fractions, the Peak LHGR is a combination of rod average power and the axial power profile maximum factors. For example, using a rod average power of 12.2 kW/ft from Figure 1 and an axial power profile maximum factor of approximately 1.15 for PWRs from Figure 2 of Ref. 24, the resulting "Peak LHGR" would be approximately 14.0 kW/ft. The maximum linear heat rate is a typical term for licensees (generally defined in safety analyses and/or Technical Specifications) and is the product of the rod average power (or linear heat generation rate determined by the core rated thermal power and the linear component of all power producing rods in the core) and the hot channel factor (FQ). These maximum linear heat rates can exceed the definition of "Peak LHGR" associated with Figure 1 of DG-1389. Without further explanation of the definition of Peak LHGR associated with the figure, misinterpretation and confusion of the Figure 1 envelope may result between the licensee and regulator.	Include the following statement in a footnote for Figure 1 or incorporated into the discussion of Section 3: From Ref. 24, the Peak LHGR is defined as the product of the peak fuel rod average power and the peak fuel rod axial power distribution. This Peak LHGR may differ from the definition used in licensees' Technical Specification or Core Operating Limits Report. A Peak LHGR derived consistent with the definition of Ref. 24 should be used for comparison of applicability to Figure 1.
17.	Section 3.2 (Release Fractions) [p.21]	The Figure 1 rod average power envelopes were derived in Ref. 24. Ref. 24 does not indicate if uncertainties were applied to the rod average powers or normalized axial power distributions in Figure 2 of Ref. 24. It is assumed that uncertainties are not applied; however, the NRC should confirm that uncertainties do not need to be considered when comparing against the bounding power profile. Note that for typical applications for determining release inventories, uncertainties may be accounted for in peaking factors such as the radial peaking factor (applied uncertainty), which is applied separately to the core inventory from the gap fractions in the radiological analyses. As such, applying uncertainties in determining the gap fractions may result in double-accounting and should not be advised.	The NRC should confirm that uncertainties do not need to be applied to the power inputs used in comparing to the bounding power profile and include this clarifying information in the discussion in Section 3.2.
18.	Section 3.3 (Timing of Release Phases) [p. 22]	Table 5 in Section 3.3 lists the duration of the gap release phase is 0.22 hours for PWRs and 0.16 hours for BWRs. However, the last sentence of the following paragraph says: "Regardless of delays in the onset, the duration of the gap release phase is 0.5 hours." The values in Table 5 are not consistent with this statement.	The NRC should delete or correct the sentence in question as it contradicts gap phase durations in Table 5.

No.	Section	Comment/Basis	Recommendation
19.	Section 3.3 [p. 22]	DG-1389 states: "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." RADTRAD models core release as a constant fraction of the core inventory over the release duration. Including decay of the core inventory will make this release slightly non-linear, decreasing over time as the core inventory decreases due to decay.	Clarify that the statement in the guidance excludes the effects of decay.
20.	Section 3.4 (Radionuclide Composition) [p. 22]	Table 6 presents the radionuclide groups which should be considered in design basis analyses. The table presents molybdenum as included in the noble metals group. This is inconsistent with Tables 1 and 2 which present molybdenum as a separate group with different release fractions than the noble metals group.	The NRC should present molybdenum as a separate group in Table 6 in order to maintain consistency with Tables 1 and 2.

21.	Section 4.2 (Control Room Dose Consequences) [p. 24]	DG-1389 includes a statement that the transit dose to personnel traveling to and from the control room should be considered for licensees whose licensing basis includes transit dose. This statement should be clarified or removed based on the following points:
		• For licensees who do not currently include transit dose in their licensing basis, will the NRC force the adoption of transit dose if this regulatory guide is adopted?
		• For licensees who do currently include transit dose in their licensing basis, will the NRC allow for removal of the transit dose based on the positions summarized below:
		  • GDC 19 identifies that "Adequate radiation protection shall be provided to permit access and occupancy of the control room…". GDC 19 does not identify that access includes the need for the control room to provide protections outside of the control room envelope. Additionally, radiological consequences analyses are intended to provide an evaluation of the design and performance of structures, systems, and components of the facility. This intent is to confirm the design, construction, and siting of the facility and applicable safety features are adequate to limit dose exposure; thus, operator actions outside of the CR envelope are not historically considered in the design basis analyses.
		  • During emergency situations, personnel dose is governed by ALARA principles such that Emergency Planning will limit the dose received by operators during transit to and from the control room. 10 CFR 50.47(a)(9), (10), and (11) provide sufficient requirements for the Emergency Response organizations to measure and limit doses to emergency response personnel including control room operators.
			The NRC should remove these statements from the RG on the basis that transit doses would be addressed by ALARA principles through Emergency Planning. These measures may include alternate travel pathways to and from the control room such that operators are not traversing the radioactive plume and personal protective equipment (e.g., respirators).

No.	Section	Comment/Basis	Recommendation
22.	Section 4.2 (Control Room Dose Consequences) [p. 24]	The occupancy factors in RG 1.183 R0 are based on RG 1.183 R0 Reference 22. Reference 22 is the paper by K. G. Murphy and K. W. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," published August 1974. In that paper, the occupancy factors are described as follows: "...**<u>an allowance may be considered for the time the operator leaves the plant vicinity</u>**. This is described as the <u>occupancy factor</u>." Table 1 in the paper by K. G. Murphy and K. W. Campe provides the occupancy factors included in the determination of the X/Q values using the K. G. Murphy and K. W. Campe methodology. Therefore, it is reasonable to conclude the occupancy factors (1, 0.6 and 0.4) assumed for the control room analysis already implicitly account for the time for the operator travel from the parking lot to the control room. The occupancy factor discussion is in the X/Q section (page 13) which is consistent with this argument. No additional discussion is required with those occupancy factors. Further the Murphy and Campe paper interprets the GDC in this way: “Whole body gamma radiation from direct shine radiation sources external to the control room and from the airborne activity within the control room should not exceed a total of 5 rem.” Direct shine here is from outside radioactivity directly to operators within the control room. This would further exclude a transit dose for the operators. As well, in NUREG-0800, Section 6.4, referring to individuals within the control room says: “In accordance with GDC 19, these doses to an individual should not be excluded for any postulated accident. The whole body gamma doses consists of the contributions from airborne radioactivity inside and outside the control room, as well as direct shine from all radiation sources.” Industry has general evaluated the other sources in terms of containment shine and filter shine.”	Remove the following discussion as it is unnecessary: “The following guidance should be used in determining the TEDE dose for demonstrating compliance with 10 CFR 50.67(b)(2)(iii). For the purpose of this RG, a transit dose is considered to be the dose that is accumulated as personnel travel to and from the control room for the duration of an accident once onsite (e.g., dose from site boundary to the control room). Licensees whose licensing basis includes transit dose should include the transit dose to demonstrate compliance with 10 CFR 50.67(b)(2)(iii). The licensee’s results for the evaluation of transit dose will be evaluated on a case-by-case basis. New reactor licensees that are required to show compliance with GDC 19 or similar control room radiological habitability principal design criteria may use this guidance.”

No.	Section	Comment/Basis	Recommendation
23.	Section 4.2.3 (Transport Models) [p. 25]	Footnote 15 states that nuclides used for modeling dose from airborne radioactivity inside the control room may not be conservative for determining the dose from radioactivity outside of the control room. This statement is vague and does not provide the licensee with details as to which nuclides the NRC is concerned about for the different control room dose aspects (inhalation/immersion versus plume shine dose). Without further explanation, it is assumed that the NRC is in agreement that the sets of nuclides used by licensees acceptably addresses the footnote.	The NRC should provide clarification for Footnote 15 in terms of which nuclides are of concern for the different aspects of the control room dose. Lack of explanation infers that the NRC accepts the licensees' currently analyzed source terms and no updates are needed.
24.	Section 4.2.7 (Dose Conversion Factor) [p. 26]	This section provides the expression to correct the semi-infinite cloud dose to a finite cloud dose for external exposure. This expression is generally incorporated into radiological consequences codes (e.g., RADTRAD) and does not specifically need to be considered as an additional factor applied to the control room dose results.	The NRC should note that the expression may be incorporated into radiological consequences codes (e.g., RADTRAD) and does not need to be specifically applied to the control room dose results.
25.	Section 4.4 (Acceptance Criteria) [p. 28]	Table 7 provides the analysis release duration for the various accidents presented in the regulatory guide. The following items should be addressed by the NRC: • The PWR Steam Generator Tube Rupture, PWR Main Steamline Break, and PWR Locked Rotor identify that the analysis release duration is "Until cold shutdown is established." For consistency with the event-specific guidance presented in Appendices E, F, and G, this wording should be changed to "Until shutdown cooling is in operation and releases from the SG(s) have been terminated." • For Fuel Handling Accident, the analysis release duration should be updated for consistency with the updated FHA model presented in Appendix B. The updated model presents releases in two phases which potentially span the entire standard 30 days of event duration.	For consistency with the event-specific guidance presented in Appendices E, F, and G, change wording from the analysis release duration is "Until cold shutdown is established" to "Until shutdown cooling is in operation and releases from the SG(s) have been terminated." In the Fuel Handling Accident, update the analysis release duration with the updated FHA model presented in Appendix B.

26.	Section 5.3 (Atmospheric Dispersion Modeling and Meteorology Assumptions) [p. 30-32]	Guidance for the modeling of the limiting X/Qs for the LPZ and control room is presented and directs the licensee to model the period of most unfavorable atmospheric dispersion factors coincident with the time period of most adverse environmental release. This guidance has been included previously for the control room location in RG 1.194 (and identified in the 2015 periodic review of RG 1.194 as an item for alignment). However, the guidance has not been previously incorporated in or identified as needed for RG 1.183. [Also, no identification of this guidance has been made for RG 1.145 for the LPZ location.] The current widely used practice for modeling X/Q values for the LPZ and control room is to align the values in correct time period order, e.g., 0-2 hrs, 2-8 hrs, etc. (similar to the first portion of Figure 2 in DG-1389). However, it is recognized that this may not align with the period of most adverse environmental release in instances of delayed releases from damaged/melted fuel (in cases of an MHA) or prolonged buildup and transit from the primary to secondary systems to the environment (in cases of a locked RCP rotor). The interpretation of the guidance presented in DG-1389 is that the period of most adverse release aligns with the period of highest dose. However, the NRC has not provided sufficient guidance for consideration of situations in which the period of most adverse environmental release does not align with the period of highest dose. This may be the case for instances in the control room depending on specific modeling assumptions. For instance, for a PWR MHA, a staged release is modeled in accordance with Table 2 of RG 1.183. This results in the limiting environmental release typically occurring after the initial 0-2 hour time period. The control room is typically isolated early in the event, accounting for some delay, such that only gap activity enters the control room envelope during unfiltered, normal ventilation mode. Therefore, during the period of greatest environmental release, the activity entering the control room would be subject to emergency mode flows (possibly lower than normal mode), filtration (~90-95% iodine and particulates), and lower X/Q values associated with emergency mode intakes (depending on the receptor positioning). As a result, applying the proposed guidance would decrease the control room dose consequences relative to the current practices. Additionally, no guidance has been provided for cases in which multiple release pathways are modeled with varying periods of limiting environmental releases and/or dose. For simplicity, and to avoid potential non-conservative applications of the guidance, it is recommended that the NRC remove this position from DG-1389 and maintain the current practice of applying LPZ and CR values from highest X/Q at event initiation (T=0) for all	The NRC should re-assess the need to include this guidance for atmospheric dispersion factor modeling. The guidance, as presented, may result in a misappropriation of the limiting two-hour X/Q values for the time period of greatest release as this may not align with the time period of greatest dose contribution for the control room. It is recommended that the NRC remove this position from DG-1389 and maintain the current practice of applying LPZ and CR values from highest X/Q at event initiation (T=0) for all events, regardless of release magnitude. It is also recommended that this position be removed from RG 1.194.

No.	Section	Comment/Basis	Recommendation
		events, regardless of release magnitude. It is also recommended that this position be removed from RG 1.194.	
27.	Appendix A, Section A-2.2, Page A-2 [p.39]	This section indicates that the aerosol deposition models in NUREG/CR-6189 are still applicable; however, based on Section 1 of NUREG/CR-6189, the models are based on the release fractions and timing in NUREG-1465. Considering the significant changes to the release fractions and timings in DG-1389, there may be significant impacts to the deposition rates in this NUREG.	Confirm the continued applicability of the NUREG/CR-6189 aerosol removal rates.
28.	Appendix A, Section A-2.5, Page A-3 [p.40]	Revise A-2.5 to allow credit for suppression pool scrubbing based on: ➢ NUREG/CR-6153 – provides models for accident dose calculations using the AST for the purposes of crediting iodine decontamination provided by suppression pools • Accounts for the changing aerosol distribution following passage through the pool ➢ State-of-the-Art Reactor Consequence Analyses (SOARCA) project results (ML20304A339- NRC Brochure) include the suppression pool in their models and indicate that all modeled accident scenarios, progress more slowly and release smaller amounts of radioactive material than calculated in earlier studies.	Change A-2.5 to reference NUREG/CR-6153 and allow credit for reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs.
29.	Appendix A, Section A-3.5, Page A-4 [p.41]	For gas-filled secondary containment bypass leakage paths, this section states "deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis." Plate-out of elemental iodine may also be a significant removal mechanism in gas-filled secondary bypass leakage pathways. In addition to aerosol deposition, the plate-out of elemental iodine should not be excluded.	Add "deposition of aerosol radioactivity and plate-out of elemental halogens …"
30.	Appendix A, A-5. Main Steam Isolation Valve Leakage in Boiling-Water Reactors, Page A-6 [p.43]	Multiple BWRs currently have credit for aerosol removal from drywell sprays as well as aerosol deposition within in the main steam lines (some also have Condenser removal) in their current licensing basis. Section A-5 presents three acceptable methods for calculating aerosol deposition within the main steam lines, but this section states, "…however, these methods are not valid if credit has been taken for aerosol removal from drywell sprays." Given the prevalence of credit for both sprays and steam line deposition, why is there not a model presented that the Staff finds acceptable for crediting both, or modifications to the presented models if the licensee wants to credit spray removal (e.g., different aerosol size distribution)?	Provide a model where it is acceptable to credit aerosol removal from drywell sprays as well as aerosol deposition within in the main steam lines.

No.	Section	Comment/Basis	Recommendation
31.	Appendix A, A-5. Main Steam Isolation Valve Leakage in Boiling-Water Reactors, Page A-6 [p.43]	What is the technical justification for drywell sprays aerosol removal not being valid if used alongside main steam isolation valve leakage? The removal mechanisms associated with spray described in NUREG/CR-5966 are largely different than the removal mechanisms associated with deposition described in SAND2008-6601 and AEB 98-03. Section 6.2 of SAND2008-6601 states that sprays should not be used in conjunction with main steam line deposition, but this discussion is based on the steam dome being the source of radioactivity rather than the accepted position of the drywell being the source of radioactivity.	Revise words to state that the impact of sprays on aerosol removal should be determined in the submittal, and do not state that removal by both sprays and deposition should not be used.
32.	Appendix A, A-5. Main Steam Isolation Valve Leakage in Boiling-Water Reactors, Page A-6 [p.43]	DG-1389 states that the reported 3 aerosol methods are not valid if credit has been taken for aerosol removal by drywell sprays. Are other aerosol removal mechanisms similarly affected? Can a licensee credit the natural removal mechanisms in NUREG/CR-6189 consistent with Section A-2.2 and apply the reported MSL models? Although the deposition models in NUREG/CR-6189 may affect the aerosol size distribution, this impact would be expected to be similar to the impact of the MSL models themselves such that the aerosol distribution entering the MSLs are not significantly different from that assumed in the models.	Confirm acceptability of applying NUREG/CR-6189 aerosol deposition in addition to reported MSL deposition models.
33.	Appendix A, A-5. Main Steam Isolation Valve Leakage in Boiling-Water Reactors, Page A-6 [p.43]	DG-1389 reports the first MSL aerosol deposition model as: *Direct adoption of the SAND 2008-6601 (Ref. A-11) recommendations without scaling "R*-factors;"* This approach needs additional explanation. As defined in Section 1.1 of SAND2008-6601, the R* factor is defined as the ratio of NUREG-1465 containment airborne concentrations to MELCOR containment airborne concentrations. A separate factor, R_M, models the ratio of the steam dome concentration to the drywell concentration determined by the MELCOR full plant analyses. Backing the R* factor out of the process does not appear to be possible since, per Section 5.2 of SAND2008-6601, the R_M and R* factors are combined and their product is applied to develop the results in Table 5-3. This section should adopt the MHA approach where the drywell and steam dome are well-mixed. As such, there should be <u>no</u> scaling factor applied to model the increased concentration in the steam dome.	Revise first aerosol deposition model to "… without scaling R_M and R* factors".

No.	Section	Comment/Basis	Recommendation
34.	Appendix A, A-5.6, Page A-9 [p.46]	It is understood that there are several deposition and removal mechanisms (see Table 4.1-3 of the State-of-the-Art Report for examples) present inside of steam lines and it is challenging to provide a comprehensive model due to lack of experimental data and disassociation between the reality of core cooling being maintained during a design basis accident and fuel damage being assumed per DG-1389. The State-of-the-Art Report attempts to capture all of these uncertainties but cannot and does not form a conclusion based on all of the available literature. Based on previous approval of AST applications and the lack of concrete knowledge surrounding removal in steam lines, it is recommended to allow licensees to continue to use their existing main steam line removal models if pursuing licensing actions unrelated to the steam lines themselves.	Change footnote 4 to state that previously approved methods are not superseded by the RG 1.183 Revision 1 methods if unaltered during submittals associated with licensing actions unrelated to steam line removal.
35.	Appendix A, A-5.6, Page A-9 [p.46]	Aerosol deposition in vertical volumes is non-zero. SAND2008-6601 Section 3.3 states that some deposition occurs in vertical surfaces. This is confirmed by Figure 2-16 and Figure 2-17 of the State-of-the-Art Report which show strong correlations between temperature driven deposition by thermophoresis and condensing vapor deposition by diffusiophoresis. These removal mechanisms are largely independent of gravitational settling in horizontal segments.	Remove “in horizontal volumes” or provide different guidance for horizontal vs. vertical volumes when applying Method 1: Direct adoption of the SAND 2008-6601 recommendations without scaling RM and R* factors.
36.	Appendix A, A-5.6, Page A-9 [p.46]	Section A-5.6 implies that only plants other than Mark I, II, or III will be considered on a case-by-case basis. All models should be considered on a case-by-case basis considering the models described by positions A-5.6.1, A-5.6.2, and A-5.6.3 do not consider all removal mechanisms present in the steam lines.	Revise statement to allow case by case review for all technologies.
37.	Appendix A, A-5.6, Page A-9 [p.46]	Section A-5.6 indicates that the multi-group method “will be evaluated on an individual case-by-case basis.” This statement implies that the application of this method may not be fully approved, thereby increasing the potential for regulatory uncertainty in future submittals. Why does the approach described in Section A-5.6.2, including an AMMD of 2.0 µm, 2,000 groups, and 10,000 trials, require additional regulatory review?	Confirm that the approaches in Sections A-5.6.2 and A-5.6.3 are acceptable models for aerosol deposition in the MSLs.

No.	Section	Comment/Basis	Recommendation
38.	Appendix A, A-5.6.1, Page A-9 [p.46]	The basis for not being able to credit the piping upstream of the inboard MSIV seems to come from Section 6.3 of SAND2008-6601 which states that "at times in the simulation the temperature of portions of the in-board MSL piping are predicted to be high enough to vaporize fission products that had been previously deposited." The MELCOR simulation in SAND2008-6601 is not representative of plant specific thermo-hydraulic conditions. (e.g., Figure 2-19 of SAND2008-6601 shows a long term temperature of approximately 800F in the steam dome while most BWR analyses would show long term temperatures of less than 300F in the core with ECCS operational). Credit should be able to be taken for deposition in the inboard lines if plant specific analysis shows low temperatures considering RIS 2006-04 states that deposition of particles in the inboard volume occurs. In addition, the time periods in DG-1389 are inconsistent with SAND2008-6601.	Revise table to be consistent with SAND2008-6601 and add in calculated removal for in-board lines.
39.	Appendix A, A-5.6.2, Page A-9 to A-11 [p.46-48]	Calculations typically consider a "single failure" which is either assumed to be a break in a steam line leading to not modeling holdup/deposition in a steam line upstream of the inboard MSIV or assumed to be a stuck open MSIV which leads to combining the inboard volume with the between MSIV volume in a single steam line. The discussion states that a pipe break is not assumed, so should all applicants consider the limiting single failure to be a stuck open MSIV (with respect to deposition in the steam lines)?	Clarify that there is no in-board pipe break in the MHA scenario.
40.	Appendix A, A-5.6.2, Page A-9 to A-11 [p.46-48]	The State-of-the-Art report recommends a AMMD of 3 μm for containment and Section A-5.6.2 describes the approved approach as applying an aerodynamic mass mean diameter (AMMD) of 2.0 μm. This appears to be an average value of the RCS and containment AMMD values of 1.0 and 3.0 μm respectively. Considering the MHA assumption of a well-mixed drywell, the containment AMMD of 3.0 μm would be the applicable parameter on which to base the aerosol size distribution entering the main steam lines.	Revise the text to apply a 3.0 μm AMMD or explain the basis for the suggested value of 2.0 μm.
41.	Appendix A, A-5.6.2 and A-5.6.3 Page A-9 to A-12 [p.46-49]	These methods are based on the AEB 98-03 methodology which does not account for thermophoresis, impaction, diffusiophoresis, flow irregularities, and hygroscopicity. The removal calculated is largely correlated to particle size but because it does not include these other mechanisms it is overly conservative. For example, Section 2.6.3 of the State-of-the-Art Report states that thermophoretic deposition velocity is not an especially strong function of particle size.	Clarify that other associated removal mechanisms may be calculated separately and included in the TEARE.

<table>
<tr><th>No.</th><th>Section</th><th>Comment/Basis</th><th>Recommendation</th></tr>
<tr><td>42.</td><td>Appendix A, A-5.6.4, Page A-12 [p.49]</td><td>Section A-5.6.4 indicates that aerosol deposition in the condenser using a multi-group or numerical integration approach needs to be evaluated on an individual case basis.

Why would the multi-group or numerical integration approaches not be applicable in the condenser? These models consider the impact of easier-to-remove particles being removed in the upstream compartments, leading to less deposition in the condenser. Can the condenser deposition coefficients in Table A-1 be applied instead of a value developed with the multi-group or numerical integration approaches?</td><td>Clarify use of Table A-1 condenser deposition coefficients or clarify why the multi-group method is not applicable to the condenser.</td></tr>
<tr><td>43.</td><td>Appendix B (Fuel Handling Accident) [p. 52]</td><td>Footnote 1 in Appendix B presents that if a postulated event (heavy load drop) occurs for an FHA, the activity release may be based on core average gap fractions presented for the LOCA in Tables 1 and 2. The following issues should be addressed by the NRC:

• Tables 1 and 2 identify gap fraction releases for the Te, Ba, and Sr groups. As identified in a previous comment, the SAND-2011-0128 report, in which Tables 1 and 2 are based, clarified that these releases listed in the gap phase were more accurately classified as in-vessel releases that occurred during the gap release phase. As a heavy load drop event is a shutdown event and fuel damage should only result from a load impact, there would not be a driver for further fuel failure associated with in-vessel releases (which results from fuel melt). Therefore, these nuclide groups should be excluded from consideration in the fuel handling accidents involving the entire core.
• The Tables 1 and 2 release fractions are associated with the timings presented in Table 5. Table 5 presents gap phase durations of 0.22 hours (PWRs) and 0.16 hours (BWRs). The timings in Table 5 are based on simulations of various plant transients which do not include fuel handling accidents / heavy load drop events. Therefore, it should be specified that the use of Tables 1 and 2 do not need to explicitly follow the gap phase durations from Table 5 and can instead use the durations presented in this appendix (Appendix B).</td><td>Replace the footnote wording with the following to clarify the gap fractions and timings to apply for the heavy load drop accident:

“These assumptions may also be used in assessing the radiological consequences of a heavy load drop over fuel accident. If the event is postulated to damage all of the rods in the core, the release activity may be based on the core-average gap fractions of Tables 1 and 2 in the main text of the guide, and the radial peaking factor may be omitted. Gap fractions may be limited to the noble gas, halogen, and alkali metal groups, consistent with Tables 3 and 4. Additionally, the release timings and durations associated with this appendix may be used in lieu of release timings and durations associated with Regulatory Position 3.2 for Tables 1 and 2.”</td></tr>
</table>

No.	Section	Comment/Basis	Recommendation
44.	Appendix B, B-1.3, Page B-1 [p. 52]	In the current licensing basis fuel handling accidents, the impacts of stable and long-lived iodine isotopes (e.g., I-127 and I-129) are not typically considered due to their negligible dose consequences. For the revised fuel handling accident approach, these isotopes are important to develop the pool iodine concentration as they represent most of the iodine inventory in the fuel rods. The NRC's assessment in the staff's example in ML21190A040 appears to apply a much larger value of 23% for Iodine-129 which is significantly higher than the 4% from Table 4 for "other halogens."	Confirm the applicability of Tables 3 and 4 for I-127 and I-129 as "other halogens."
45.	Section B.1-3	DG-1389 only reports the iodine species distribution released into the pool water. Reg Guide 1.183 Rev. 0 reported that the iodine chemical species above the water is 57% elemental and 43% organic. For Rev. 1 applications, what iodine species distribution should be applied for the early 2-hour airborne release considering the new fuel handling accident model? Based on the calculated overall pool DF from Equations 3-1, 3-2, and 3-3, is a species-dependent DF approach acceptable with organic iodine having a DF of 1? Is a purely elemental iodine species appropriate for the long-term release since this is comprised of re-evolved elemental iodine?	Provide additional guidance on acceptable airborne iodine species assumptions.
46.	Appendix B, B-2, Page B-2 [p. 52]	There can be fuel handling accident scenarios where the water depth may be greater than 23 feet such as drops over the core. However, Section B-2 states that the DF can be calculated from Equations in B-1, B-2, and B-3, if the water depth is between 19 and 23 feet. There is no guidance in the event the water depth is greater than 23 feet. Since the DG-1389 equations are not dependent on depth, it would be expected that they would yield a conservatively low DF than an actual value for cases that credit the additional scrubbing depth.	Revise statement as follows: "For water depths greater than or equal to 19 feet, an overall iodine DF based on pin pressure is computed as follows: …"
47.	Appendix B-3 (Phase 2 Release – Re-evolution Release) [p. 54, B-3]	The mass transfer coefficient (KL) is presented as a constant value (3.66x10-6 m/s). From Enclosure 4 of Reference B-1, this is a combined liquid-gas phase coefficient that does not consider recirculation in the pool and is based on the assumption of a high flow rate to clear the building air.	The NRC should clarify that the mass transfer coefficient remains applicable for all conditions (i.e., recirculation in the pool and low flow rate to clear the building air) and no adjustments need to be made for changing conditions.

No.	Section	Comment/Basis	Recommendation
48.	Appendix B-3 (Phase 2 Release – Re-evolution Release) [p. 54, B-3]	Step 2 of the FHA model instructs the analyst to calculate the amount of iodine in the fuel pin gap. The calculations consider I-131 for the amount of radioactive iodine (with subsequent clarification to include other radioactive iodine isotopes to the total) and consider I-129 for the amount of non-radioactive iodine (with no subsequent clarification to include other non-radioactive iodine isotopes). The radioactive and non-radioactive iodines are then combined for a total iodine which is used in the downstream calculations. [Note that the individual components are not used in any downstream calculations within the model and calculations involving only the radioactivity iodine can be removed from the model development.] It was observed in the calculations that a higher total iodine amount is conservative for calculating the release rate to be used in the analysis. As such, additional non-radioactive iodine isotopes (I-127) should conservatively be considered.	The NRC should specify that additional non-radioactive iodine isotopes (especially I-127) should be considered to maximize the amount of iodine considered in Step 2 of the FHA model calculation.
49.	Appendix B-3 (Phase 2 Release – Re-evolution Release) [p. 55, B-4]	Equation B-10 in Step 2 calculates the [I2]/[I-]2 concentration ratio using the following equation: $R_i = [I_2]/[I\text{-}]^2 = C_h^2 \,/\, (6.0603E\text{-}14 + 1.4708E\text{-}09\ C_h)($ Constants 6.0603E-14 and 1.4708E-09 originate from Ref. B-2 (NUREG/CR-5950); however, the constants are not presented in the reference to that degree of accuracy. The constants presented in NUREG/CR-5950 are 6.05E-14 and 1.47E-09 [NUREG/CR-5950 page 13 and Appendix C, page C.3]. It is unclear where the discrepancy is occurring.	The NRC should provide clarification as to the discrepancy in constants between NUREG/CR-5950 and DG-1389 or update Equation B-10 to present the constants to the degree of precision from NUREG/CR-5950.
50.	Appendix B-5 (Fuel Handling Accidents within the Fuel Building) [p. 56, B-5]	Position B-5.1 provides a release duration time for the first phase releases (2 hours). The position does not provide any specifics for onset and duration timing for the second phase releases (re-evolution releases). From the Ref. B-1 model background, this release phase should begin immediately after the first phase (2 hours) and be modeled until the typical end of the radiological considerations (30 days). It is assumed that there is no overlap of release phases. Note that this comment is also applicable to Position B-6.3.	The NRC should clarify the starting time and duration of the re-evolution phase.
51.	Appendix B-5 (Fuel Handling Accidents within Containment) [p. 56-57, B-5-6]	Footnote 4 breaks across pages B-5 and B-6. This footnote should be contained on a single page. Note that this footnote is referenced in Positions B-6.1 and B-6.2 which are currently on separate pages, so Footnote 4 may have to be repeated on separate pages. Likewise, Footnote 3 for Position B-6.4 is not located on the same page as the position.	See comment for the items to address.

No.	Section	Comment/Basis	Recommendation
52.	Appendix E (Steam Generator Tube Rupture) [p. 63]	Revision 0 of RG 1.183 presented the PWR Main Steamline Break event in Appendix E. There is no indication as to the motive behind the switch in appendices for DG-1389. This may cause confusion to licensees which need to compare/contrast the RG revisions or those that elect to implement and maintain different RG revision methods for these events.	The NRC should consider maintaining consistency in the event order presented in the regulatory guide appendices.
53.	Appendix E (Steam Generator Tube Rupture) [p. 63]	The page numbering for Appendix E is not correct as the first page of the appendix is identified as page E-3.	The NRC should correct the page number in Appendix E and also review the entirety of the regulatory guide to correct any further editorial errors.
54.	Appendix E-6 (Transport - Steam Generator Tube Rupture) [p. 64]	Position E-6.1 states that the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate for the limiting condition for operation specified in the TSs. This leak rate is associated with significant pressure differentials across the steam generator tubes which typically occur early in the transient. In later periods of the transient, and especially during longer-term steaming for cooldown, the pressure differential is significantly decreased. Licensees should be able to credit this significant reduction in leakage if it is shown to be credible. This comment is also applicable for other steam release events which model primary-to-secondary leakage. [MSLB in Appendix F, Locked Rotor in Appendix G, and Rod Ejection in Appendix H]	The NRC should add a statement that licensees may credit a reduction in the TS leakage rates for later periods of the transient when pressure differentials across the steam generator tubes are significantly reduced based on adequate technical justifications.

No.	Section	Comment/Basis	Recommendation
55.	Appendix E-6 (Transport - Steam Generator Tube Rupture) [p. 66]	Position E-6.6 identifies that the potential impact of tube uncovery on the transport model parameters needs to be considered. The issue of tube uncovery was addressed by the Westinghouse Owners Group (WOG) in WCAP-13247, "Report on the Methodology for the Resolution of the Steam Generator Tube Uncovery Issue," March 1992. The WOG program concluded that the effect of tube uncovery would be essentially negligible and the issue could be closed without any further investigation or generic restrictions. This position was accepted by the NRC in a letter dated March 10, 1993, from Robert C. Jones, Chief of the Reactor System Branch, to Lawrence A. Walsh, Chairman of the WOG. The letter states " ... the Westinghouse analyses demonstrate that the effects of partial steam generator tube uncovery on the iodine release for SGTR and non-SGTR events is negligible. Therefore, we agree with your position on this matter and consider this issue to be resolved." Consistent with this position, tube uncovery should not need to be addressed further for U-tube style steam generators. This comment also applies to Positions F-6.6.3 for MSLB, G-5.5 for Locked Rotor, and H-7.4 for Rod Ejection.	The NRC should update the wording of this position to acknowledge for U-tube steam generators that the issue of short-term tube uncovery has previously been resolved and does not need to be considered in future applications.
56.	Appendix I [p. 76]	Appendix I is not referenced within DG-1389 and the purpose and importance of this appendix is not understood. Given the unspecified use and importance of this appendix, it is recommended the NRC remove the appendix and flowchart.	NRC should delete this appendix or provide reference in document as to why this is needed.
57.	Appendix J [p. 77]	The first paragraph of this appendix cross-references Section C of the main body of this guide for the release fractions in Tables 3 and 4. This is an editorial error as Tables 3 and 4 are located in Section 3.2.	The NRC should correct this cross-reference and review DG-1389 for any other potential cross-referencing errors. Section C position 3.2.