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6

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Rulemaking and Directives Branch
Division of Administrative Services
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

Subject: Nuclear Energy Institute Comments on U.S. Nuclear Regulatory Commission Draft Regulatory Guide DG-1199, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (*Federal Register* of October 14, 2009, 74 FR 52822).

Project Number: 689

This letter provides comments of the Nuclear Energy Institute (NEI)[1] on behalf of the nuclear energy industry on U.S. Nuclear Regulatory Commission (NRC) draft regulatory guide DG-1199 (proposed Revision 1 to NRC Regulatory Guide 1.183), in response to the subject Federal Register notice.

These comments were developed by a nuclear energy industry task force made up of subject matter experts from 18 companies involved in the use of alternative radiological source terms for evaluating postulated design basis accidents (DBAs) at nuclear power reactors. The task force comments reflect a substantial body of industry licensing and technical expertise, experience, and lessons-learned gained from successful licensing actions utilizing alternative radiological source terms as part of the supporting DBA analyses.

We are aware that the Boiling Water Reactor Owners' Group (BWROG), in a letter to the NRC dated January 6, 2010, proposed that the NRC extend the period for receiving comments on DG-1199 and delay final issuance of Revision 1 to Regulatory Guide 1.183 to allow time for the BWROG to

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[1] NEI is the organization responsible for establishing unified nuclear industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all utilities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel fabrication facilities, materials licensees, and other organizations and individuals involved in the nuclear energy industry.

SUNSI Review Complete

E-RIDS= ADM-03

Template = ADM-013

Add = R. Carpenter (rec1)
M. Blumberg (wmb1)

"perform a detailed review of the Staff research supporting the proposed changes to the modeling of main steam isolation valve (MSIV) leakage." We fully endorse and support the BWROG request.

In addition to the rationale provided by the BWROG for delaying final issuance of the guide, we also note that draft regulatory guide DG-1199 has been issued for public comment while some supporting documents that are needed to complete our review of the draft guide have not yet been made publicly available. In a public workshop held at NRC headquarters on November 16, 2009, NRC staff referred to "adopting the ANS 5.4 (2009) model," as the technical basis for revised gap release information in DG-1199. It is our understanding that ANS 5.4 is still in draft form and has not yet been entered into the final consensus balloting process, during which time the draft standard itself will be subject to review and comment and possible further revision. NRC staff also referred to NUREG/CR-XXXX as providing the "technical basis for derivation of [the] revised ANS 5.4 release model." It is our understanding that the NUREG is in publication, but has not yet been issued as a publicly available document. In addition, NRC states in DG-1199 that Regulatory Guide 1.89 on environmental qualification of certain electrical equipment is undergoing revision and will soon be published for review and comment as DG-1239. In our view, the full impact of the proposed changes in DG-1199 cannot be determined until DG-1239 is publicly available.

In addition to supporting the BWROG request, we recommend that the NRC defer issuing Revision 1 to Regulatory Guide 1.183 until all of the supporting and related documents are publicly available for concurrent review and any additional comments resulting from that review have been considered by NRC staff in the final version of the guide.

Such a deferral will also allow time for NRC staff to better clarify its intent in regard to licensee implementation of the regulatory guide after it is issued. DG-1199 includes the statement, "NRC staff will use this guide to evaluate licensee-initiated changes if there is a clear nexus between the proposed change and the guidance contained in the guide." This statement is inconsistent with another statement in the draft guide that "this regulatory revision provides licensees with an opportunity to use an updated method for determining control room and offsite radiological assessments, *if that is the method the licensee prefers*" [emphasis added]. In order for stakeholder comments to fully reflect an understanding of the potential burden and benefits associated with the proposed "updated method," it is necessary to clearly understand how the guide will apply to future licensing actions, particularly for the 81 licensees who have already fully or partially incorporated an alternative radiological source into their licensing basis and the 9 licensees who currently have such changes pending.

Notwithstanding the BWROG and NEI requests for deferring issuance of the guide, we are providing detailed comments on DG-1199 (Attachment 1). We are also providing responses to the four questions posed by NRC staff in the subject Federal Register notice (Attachment 2). We intend to submit additional comments on DG-1199 after the BWROG has completed the analysis described in its letter, and after the aforementioned supporting and related documents become publicly available.

Due to the extensive nature and significance of the changes being proposed in DG-1199, we would like to meet with NRC staff to discuss our comments, as well as to better understand NRC expectations in regard to licensee implementation of the final guide. We suggest that such a meeting should occur after the NRC staff has been able to review all of the stakeholder comments on the proposed changes.

If you have any questions regarding our comments, please contact me at 202.739.8111; rla@nei.org.

Sincerely,

Ralph Andersen

Ralph L. Andersen

Attachments

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1		General Comment		COMMENT The draft regulatory guide, DG-1199, has been issued for review and comment while other supporting documents are not yet publicly available, e.g., NUREG CR/XXXX and ANSI 5.4 (that form the technical basis for updates for the new proposed Non-LOCA gap release fractions), and other documents are either currently undergoing revision or will need to be revised as a result of the proposed changes in DG-1199, e.g., NUREG-1465, Regulatory Guide 1.89, and Standard Review Plan 15.01. This fragmented approach runs contrary to the agency's efforts to improve openness and transparency. RECOMMENDATION Proposed Revision 1 to Regulatory Guide 1.183 should not be finalized until all documents related to draft regulatory guide DG-1199 have been made available for concurrent review and comment by stakeholders, including providing supplemental comments on DG-1199. Comments being provided on DG-1199 should be used to inform additional changes to the draft guide, as well as to inform proposed changes to other documents that are being or will be revised in conjunction with DG-1199. In addition, further comments on DG-1199 should be actively sought and considered when all of the other related documents become publicly available, including final supporting documents, such as NUREG CR/XXXX and ANSI 5.4, and draft

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				revised documents, such as NUREG-1465, Regulatory Guide 1.89, and Standard Review Plan 15.0.1.
2		General Comment		<u>COMMENT</u> Revision 1 to RG 1.183 should be prepared with the intent of superseding RIS 2006-04. Item 9 from the RIS with respect to source terms for certain non-LOCA accidents should not be included. This item confuses guidance on chemical form with guidance on source terms. No justification other than semantics appears to support this RIS statement. <u>RECOMMENDATION</u> Upon RG 1.183 Rev. 1 completion, RIS 2006-04 should be either deleted or amended if a continued need for the RIS is identified.
3		General Comment		<u>COMMENT</u> In RG 1.195 the NRC has written guidelines for radiological analysis of design basis accidents at those plants for which the NRC has not approved full scope implementation of AST. Many of the changes proposed to the guidelines in RG 1.183 for AST analyses apply to RG 1.195 (i.e., do not apply only to an AST analysis). To date, no draft revision to RG 1.195 has been released for review and comment. <u>RECOMMENDATION</u> Revise RG 1.183 and RG 1.195 concurrently, allowing for public comment on both draft revisions prior to finalizing either document.
4		General		<u>COMMENT:</u>

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		Comment		It is understood that a revision to NUREG-1465, a basis document for RG 1.183, is in preparation. If RG 1.183 is revised before NUREG.1465, the revision to NUREG-1465 could require a second revision to RG 1.183. <u>RECOMMENDATION:</u> RG 1.183 should not be revised before NUREG-1465. The staff should at least issue the revision to both documents at the same time.
5	1	A	'This guide establishes **the** AST....'	<u>COMMENT</u>: RG 1.183 Rev 0 July 2000 Section A-Introduction states - 'This guide establishes **an acceptable** alternative source term (AST)..." <u>RECOMMENDATION</u>: Clarify if the proposed revision supersedes RG 1.183 Rev 0. Identify the applicability for DG-1199 (when issued as a final guide) relative to future licensee actions, such as power uprate, plant life extension, new plant license applications, etc.
6	2	A	The second sentence of the second paragraph states; ***"A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based upon a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products."*** However, the words above quoted from 10 CFR 100.11 are not complete. The full version of this section (titled "Determination of exclusion area, low population zone, and population center distance.") is	<u>COMMENT:</u> The words left out of the DG-1199 statement include: *"... hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible."* Removal of these words could possibly lead one to a different conclusion, thereby causing

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			as follows. ***"The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."***	confusion. RECOMMENDATION: All words need to be included as shown.
7	2	A	Footnote #2 states: **"As defined in 10 CFR 50.2, "Definitions," "design bases" means information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state-of-the-art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculation or experiments or both) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. The NRC considers the accident source term to be an integral part of the design basis because it sets forth specific values (or a range of values) for controlling parameters that constitute reference bounds for design."**	COMMENT: The footnote allows licensees to credit qualified emergency systems, structures, and components with specifically identified functional goals. However, the source term to be used in the analyses must be of a significant quantity in order to test these systems, structures, and components. This directly opposes the work performed in the Sandia MELCOR report. RECOMMENDATION: No changes are required to this section; however, it directly opposes the work performed in the Sandia MELCOR report. Credit for properly engineered and qualified safety systems, components, or structures need to be allowed in accordance with the footnote as long as an appropriate non-mechanistic source term is used.
8	2	A	Last paragraph of this page makes reference to Chapter 15.0.1 of the SRP.	QUESTION: Since several sections of SRP 15.0.1 will

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				require changes based on the DG-1199 changes, will the SRP be revised as well? If so, when will this revision be issued? <u>RECOMMENDATION:</u> Provide revisions to SRP 15.0.1 for review prior to finalizing DG-1199.
9	6	B	The first paragraph states; ***"An accident source term is intended to be representative of a major accident involving significant core damage, not exceeded by that from any accident considered credible, and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA)."*** It further states; ***"Facility-analyzed DBAs are not intended to be actual event sequences; rather, they are intended to be surrogates to enable deterministic evaluation of the response of engineered safety features (ESFs)."***	<u>QUESTION:</u> The statement that DBAs were intended to be surrogates to enable deterministic evaluation of the response of engineered safety features (ESFs) implies that the deterministic evaluation of the functions of these ESFs needs to be made. Unless these ESF functions can be credited as designed, their response cannot be evaluated. <u>RECOMMENDATION:</u> The Sandia MELCOR report appears flawed in that it needs to allow credit for qualified ESF systems and functions using an assumed severe accident source term (perhaps the one recalculated using MELCOR).
10	6	B		<u>COMMENT</u>: The statement that 'this guidance supersedes corresponding radiological analysis assumptions…when used with an approved AST" implies that plants currently using AST will now be subject to DG-1199. <u>RECOMMENDATION</u>: Clarify the applicability of DG-1199 (when issued as a final guide).
11	7	B	"For plants licensed using the ***TID-14844 source term that have not implemented an AST for EQ,*** the guidance in Regulatory Guide 1.89, Revision 1,	<u>COMMENT</u> This statement appears to focus only on plants licensed to TID 14844 source terms.

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			"Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant, Revision 1" (Ref. 11) remains valid for the determination of integrated doses for EQ purposes."	However, as discussed below, there are many plants that have implemented a full AST but continue to use TID-14844 source terms for EQ. Sec 1.3.5 of RG 1.183, Rev 0, noted that radiation environments for EQ were required to be updated to reflect the modification, but did not have to be updated simply to address the increase in Cs. Based on the conclusions of the follow-up GSI referenced in this section, the NRC took the position that from an EQ standpoint, both TID-14844 and RG 1.183 source terms are conservative in their own right. Based on this conclusion, NRC has approved many full implementation AST applications which continue to retain a TID 14844 source term as the licensing basis for EQ. At the 11/16/09 workshop with the industry, NRC indicated that there is no intent to change the above position on EQ. <u>RECOMMENDATION</u> Suggest changing the text to read: "For plants licensed using the TID-14844 source term <u>for EQ</u>, or **<u>an AST without implementing the</u>**~~that have not implemented an~~ AST for EQ, the guidance in Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant, Revision 1" (Ref. 11) remains valid for the determination of integrated doses for EQ purposes."

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12	7	C.1.1	The last sentence on the page states; ***"The NRC staff will approve these license amendment requests if the facility, as modified, will continue to provide sufficient safety margins with adequate defense in depth to address unanticipated events and to compensate for uncertainties in accident progression and analysis assumptions and parameter inputs."***	QUESTION: It is unclear as to how much defense in depth is really acceptable or even necessary. RECOMMENDATION: Provide better guidance regarding various "compensations" for consideration. A list of all assumed conservatisms should be provided such that an appropriate level can be maintained. The words "sufficient" and "adequate" are not clear in this context, adding the potential for additional RAI questions.
13	8	C.1.1.1	The first two sentences state; ***"Licensees should evaluate the proposed uses of this guide and the associated proposed facility modifications and changes to procedures to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. The safety margins are products of specific values and limits contained in the technical specifications (which cannot be changed without NRC approval) and other values, such as assumed accident or transient initial conditions or assumed safety system response times."***	COMMENT: The statement indicates that assumed safety system response times need to be considered in the analyses. RECOMMENDATION: The Sandia MELCOR report needs to be re-evaluated to allow credit for appropriately designed and qualified ESF functions. The RADTRAD analyses can still be performed using an assumed non-mechanistic severe accident source term as it is currently done. It is recommended that this analysis be re-performed. However, if this work is to be performed by industry, it will most likely not be funded, approved, or concluded before the end of the current public comment period. Therefore, additional technical comments addressing errors will be available for a future revision. Analysis results may also be submitted as a topical report available to participating BWROG members, useable for future submittals. This would require additional time for NRC reviews and the

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				potential for additional RAI questions. One reason for performing this revision is to streamline the review process, minimizing such issues. This detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes to the Sandia MELCOR report.
14	8	C.1.1.2	The first paragraph states; ***"Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties."***	COMMENT: This section requires licensees to consider system redundancy, independence, and diversity. Given ESF system redundancy, independence, and diversity as described in current licensing bases, it is unrealistic to assume that there would be no ECCS injection for the first two hours following a LOCA. RECOMMENDATION: The Sandia MELCOR report appears flawed and should be re-evaluated to allow credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public comment period for DG-1199. Additional technical comments will be available for a future revision to the regulatory guide.
15	9	C.1.1.5	This is a new section that discusses the "voluntary	QUESTION:

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			use" of RG 1.183 for new reactors.	Will new reactors have a choice as to which RG they can use? This section is not clear regarding the use of RG 1.183 for new reactors. (Also see comments for Section 1.2 below.) <u>RECOMMENDATION:</u> Provide clearer wording with respect to the NRC's intent of this section.
16	9	C.1.1.5		<u>COMMENT</u>: This section states the staff will use, where applicable, the methodology and assumptions stated in this draft revision to Regulatory Guide 1.183. <u>RECOMMENDATION</u>: Clarify that the guide will not be applicable for the review of licensee amendments and applications that are submitted before the issuance of the revised regulatory guide.
17	10	C.1.2	The 3rd paragraph specifically says that "***Full and selective implementations, as used in the regulatory positions that follow, are not applicable to new reactor applications.***"	<u>QUESTION:</u> Since Section 1.1.5 discusses the voluntary use of RG 1.183 for new reactors, it does not specifically say that AST is mandatory, why exclude these definitions? Can they not be used for new reactors? <u>RECOMMENDATION:</u> If there are only two types of implementation described in the DG ("full" and "selective") and neither is applicable to new reactor applications, provide the type of implementation required for new reactors.
18	10	C.1.2.1	"In performing this analysis, licensees should ***evaluate the spectrum of DBA LOCAs in order to ensure the bounding LOCA is identified*** and evaluated	<u>COMMENT</u> This sentence implies that licensees need to evaluate various break sizes / fuel damage

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			from a dose consequences perspective.	scenarios. It is also inconsistent with the 2nd paragraph of Appendix A that states that with respect to radiological consequences, a large break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment for siting purposes. <u>RECOMMENDATION</u> This sentence should be deleted.
19	10	C.1.2.1		RECOMMENDATION:Clarify what is meant by "evaluate the spectrum of DBA LOCAs."
20	10	C.1.2.1 Pg 14, C.1.3.4, Pg 17 C.3.3	".... a change from an approved AST to a different AST that is not approved for use at that facility would require a license amendment under 10 CFR50.67." ".... and changes to previously approved AST characteristics, requires prior NRC staff approval under 10 CFR 50.67." "After the NRC staff has approved an implementation of an AST, subsequent changes to the AST will require NRC staff review under 10 CFR 50.67."	<u>COMMENT</u> These statements imply that for a licensee that has already been approved for AST based on RG 1.183 Rev 0, implementation of Rev 1 as related to the change in the gap fractions (which is technically a change to the source terms), will require a license amendment. <u>RECOMMENDATION</u> NRC should clarify the intent of these sections.
21	10	C.1.2.2	The second sentence states; ***"The NRC staff will allow licensees to have flexibility in adopting technically justified selective implementations, provided a clear, logical, and consistent design basis is maintained."***	<u>COMMENT:</u> This section discusses the requirement for a "*technically justified selective implementations, provided a clear, logical, and consistent design basis*" for licensees. However, the same "*technically justified selective implementations, provided a clear, logical, and consistent design basis*" is not presented in the Sandia MELCOR report. <u>RECOMMENDATION:</u>

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				The Sandia MELCOR report appears flawed and should be re-evaluated to allow credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public comment period for DG-1199. Additional technical comments will be available for a future Regulatory Guide revision.
22	11	C.1.3.1	10 CFR Part 52 cited in the last bullet.	QUESTION: Is this one of the "*may include, but are not limited to*" items? The applicability to new reactors is at question. RECOMMENDATION: Provide better clarity with respect to the NRC's intent with respect to new reactors.
23	12	C.1.3.2	The first paragraph states; ***"The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and design bases appropriately. The NRC considers an analysis to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions***	QUESTION: Although this has not changed from Revision 0, it now has a different effect with regard to EQ doses. Does this now require updated EQ doses to be calculated since, although conservative, they will no longer "be valid"? RECOMMENDATION: Provide more explicit guidance (and perhaps examples) with respect to requirements for EQ doses.

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			drawn on those results, are no longer valid."	
24	12 et.al.	C.1.3.2 1st and 3rd paragraphs, C1.3.4 1st paragraph, and C.5.2 1st paragraph.	In the passages cited here, the staff writes that licensees should "update other analyses with the guidance [for AST analyses]" at least an as needed (e.g., in assessing the impact of a plant modification to equipment credited in one or more AST analyses). In addition, the NRC staff states that "Reevaluation [of a plant modification] using the previously approved source term may not be appropriate." This presents a problem in that the NRC guidelines for AST analyses do not cover all possible DBEs. The staff has written concerning this, stating the following: "The DBAs addressed ... [in the appendices to RG 1.183] were selected from accidents that may involve damage to irradiated fuel. This guide does not address DBAs with radiological consequences based on technical specification reactor or secondary coolant-specific activities only [denoted in this remark and elsewhere as the Dose Equivalent Iodine-131 or DEI DBAs]." Yet the staff further writes: "The inclusion or exclusion of a particular DBA in this guide should not be interpreted as indicating that an analysis of that DBA is required or not required. Licensees should analyze the DBAs that are affected by the specific proposed applications of an AST."	COMMENT: Taken together, these statements present licensees with a problem in their efforts to complete the implementation of AST for their nuclear plant(s). Specifically, licensees have to complete AST analyses of DBAs with no guidance at all or with guidance that are not associated with the method of AST and may include guidelines not compatible with the method of AST. This makes it all the more difficult to evaluate UFSAR updates based on AST analyses of these DBAs against the criteria of 10 CFR 50.59. The following situations illustrate the problems identified in this remark. 1) One set of DBAs each of which presumably is followed by releases of fission products to the environment but not by fuel damage are the DEI DBAs (as seen in the quote cited in this remark). The NRC staff (contrary to the stated limitation in § 5.2) has provided a set of guidelines for analysis of two of the DEI DBAs: the main steam line break (MSLB) and steam generator (SG) tube rupture (SGTR). However, the staff (this time pursuant to the stated limitation) provides no set of guidelines for an AST analysis of the third DEI DBA: "break of a letdown line or other small line carrying reactor coolant outside containment" (denoted henceforth as the interfacing line break or ILB). With

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				the AST guidelines replacing contemporary NRC expectations in the Standard Review Plan (SRP - Ref. 4) including § 15.6.2, licensees of nuclear plants for which the ILB is a DBA have no set of guidelines for an AST analysis of it. This is particularly a matter of concern for the ILB with an accident initiated iodine spike since no guidelines are in place for setting the multiplier for it. The NRC has endorsed setting the accident initiated iodine spike multiplier to 335 for an AST analysis of the SGTR and 500 for an AST analysis of the MSLB. Evaluations supporting taking 335 for the accident initiated spike multiplier for the SGTR (Ref. 5) can be used as the basis for taking this value in an AST analysis of the ILB. Other concerns are associated with the absence of guidelines for an AST analysis of ILB scenarios. One concern has to do with whether ILB scenarios with pre-existent iodine spike should be analyzed and if yes what are the acceptance criteria for offsite radiation doses for these scenarios. Another concern has to do with the duration of the accident initiated iodine spike. 2) No guidelines are in place for AST analyses of radioactivity releases from postulated radwaste tank failures. Particular matters of concern include lack of guidance pertaining to the source terms

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				and acceptance criteria for offsite radiation doses. 3) No guidelines are in place for an AST analysis of the dry storage cask drop (DSCD). The NRC has written in Interim Staff Guide-5 (ISG-5 - Ref. 6) a set of guidelines for an analysis of radiological consequences of a DSCD. These guidelines were written to endorse a method for conformance to the germane regulations in 10 CFR § 72.104(a) and § 72.106(b). As such, they are not completely appropriate for either an AST analysis of the DSCD or conformance with the germane regulations in 10 CFR 50 (10 CFR 50.67 and Appendix A General Design Criterion - GDC - 19). Situations illustrating this are provided below. The source terms endorsed in the guidelines of ISG-5 is not contained in the non LOCA source term for RG 1.183. The ISG-5 source term includes ^{3}H, ^{103}Ru, ^{106}Ru, ^{89}Sr, ^{90}Sr, ^{60}Co, and actinides that are not in the source term endorsed in RG 1.183 for non LOCA accidents. In particular, these isotopes are not in the source term endorsed in RG 1.183 Appendix B § 1.2 for AST analyses of fuel handling accidents (FHAs). It is noted that the NRC had provided guidelines for the calculation of radiation doses for the DSCD (Ref. 5 § 15.7.5) for inclusion in the UFSAR. These guidelines endorsed taking the same source terms for the FHAs

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				in the calculation of radiation doses for the DSCD (cf. Ref. 9). Like the non LOCA source term of RG 1.183 Table 3, this source term did not include the above listed radioisotopes. Several acceptance limits are set in 10 CFR 72.106(b) for radiation doses at "the nearest boundary of the controlled area" (taken to be equivalent to the Exclusion Area Boundary - EAB). Limits are set not only for the total effective dose equivalent (TEDE), but also for radiation doses to the organs, lens of the eye, and skin. This is at variance with the method of AST which includes a limit only for TEDEs. The limit of 5 Rem set for the TEDE at the nearest controlled area boundary is not equivalent to the limit for the TEDE at the EAB for any DBA. On the other hand, in the guidelines for a "pre-AST" analysis of the DSCD, the NRC endorsed setting the offsite acceptance criteria to "well within the exposure guideline values of 10 CFR Part 100, paragraph 11." For an AST analysis this equates to 6.3 Rem. ISG-5 provides guidelines for calculating radiation doses for a person at "the nearest boundary of the controlled area" for 30 days. Yet the guidelines for a pre-AST analysis of the DSCD for inclusion in the UFSAR supported the calculation of a set of 2 hour radiation doses at the EAB. ISG-5 presents no guidelines for the

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				calculation of post DSCD radiation dose in the control room. The associated regulation, 10 CFR 72.106(b) does not set a limit for post DSCD radiation dose in the control room. These situations illustrate the need for guidance for AST analyses of DBAs not covered in RG 1.183. Lack of guidance for AST analyses of these accidents may present problems in placing reports of these analyses in updates to the UFSAR under 10 CFR 50.59. <u>RECOMMENDATION:</u> RG 1.183 should include guidelines for AST analyses of all design basis accidents presumably followed by release of radioactive isotopes to the environment. Some of these DBAs may not be in the current license basis of some nuclear plants. If the staff revises RG 1.183 to include guidelines for these DBAs, they should state that if a DBA is not in the current license basis for a nuclear plant, then implementation of AST does not impose expectations for the licensee of that plant to provide an AST for that DBA. As an alternative, the staff should include in RG 1.183 a list of regulatory guides, SRP review expectations, or any other guidance documents from the NRC. The other guidance documents in this list should cover all DBAs not covered in RG 1.183. The guidelines in all of these guidance documents should be consistent with AST analyses of the DBAs not

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				covered in RG 1.183. For all cases, it is recommended that the staff refer to the guidelines in RG 1.183 § 4.2 for the calculation of post accident radiation doses in the control room. Additional recommendations are presented below to address the specific remarks. 1) Specific recommendations are made for the ILB as follows: The Staff should endorse taking 335 for the multiplier of the equilibrium iodine production rate for ILB scenarios with the accident initiated iodine spike. In RG 1.183 Appendix F § 2.2 the NRC has endorsed setting the iodine spike multiplier to 335 for SGTR scenarios. The technical basis for that endorsement (Ref. 4) justifies the same value for ILB scenarios. Also, the NRC should state that for the ILB "the assumed iodine spike duration is eight hours" (RG 1.183 Appendix F § 2.2). Finally, the staff guidelines should indicate whether ILB scenarios with pre-existent iodine spikes should be analyzed and if yes, state the offsite acceptance criteria. 2) NRC guidelines pertaining to AST analyses of radioactivity releases to the atmosphere from tank failures should list acceptance criteria for offsite TEDEs. The guidelines also should state that radioactive source term currently in the UFSAR for this event may be used in the AST analyses.

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				NRC guidelines pertaining to the DSCD should reconcile all differences between the two guidelines for "pre-AST" analyses of this DBA. Guidelines for the radioactive source term and acceptance criteria for offsite TEDEs for the DSCD should be stated. The guidelines also should resolve the above noted discrepancy in the time span for the EAB radiation dose. Finally, the NRC should determine whether for the EAB or in the control room any radiation dose other than the TEDE need be calculated for the DSCD. In making this determination, the staff should note that they have endorsed the calculation of TEDEs only for all DBAs covered in RG 1.183.
25	13	C.1.3.3	The first sentence states; **"It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated."**	COMMENT: This statement is in conflict with the last part of section 1.3.2 "affected" analysis (see Comment #16 above). RECOMMENDATION: Provide clarity with respect to the difference between affected analyses and those with "sufficient margin."
26	13	C.1.3.4	This section states; ***"After the implementation is complete, there may be a subsequent need (e.g., a planned facility modification) to revise these analyses or to perform new analyses. For these recalculations, the NRC staff expects that all characteristics of the AST and the TEDE criteria incorporated into the design basis will be addressed in all affected analyses on an individual as-needed basis."***	COMMENT: The second sentence opens up the high probability for RAI questions based on this vague mention of "as needed." RECOMMENDATION: Clarify what the as-needed bases may be, perhaps by providing some examples.

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27	13	C.1.3.4, 1st paragraph, last sentence, D, Page 34, Conclusion of the section "Regulatory Analysis."	In § C.1.3.4 the NRC staff states "Since the AST and TEDE criteria are part of the approved design basis for the facility, use of AST and TEDE criteria in new applications at the facility does not constitute a change in analysis methodology that would require NRC approval." In the conclusion of the "Regulatory Analysis, the staff further writes that "Current licensees may elect to use the updated guidance on a voluntary basis."	COMMENT: The NRC staff has approved full scope implementation of the method of AST at a nuclear plant based on its review of the AST analysis of the LOCA in conformance to RG 1.183 Rev 0. Subsequently the licensee has revised the analyses of all his DBEs with the method of AST and in conformance to RG 1.183 Rev 0 and has updated his UFSAR. Now the licensee is considering revising the AST analysis to conform to RG 1.183 Rev 1. Finally, assume that there are no attendant issues that require a license amendment request (LAR). It is not clear whether licensee can upgrade the AST analysis per 10 CFR 50.59 or if he should submit a LAR. If he submits a LAR, then by the guidelines of RG 1.183 Rev 0 and Rev 1 (§ C.1.2.1), he need submit an AST analysis only of the LOCA. However, if the reactor(s) at his plant is a PWR (are PWRs), the AST analysis of the LOCA submitted is unlikely to be different from the analysis already reviewed by the staff. Finally, in regard to the AST analysis of non LOCA DBEs the methodology in RG 1.183 Rev 1 may be conservative compared to the methodology of RG 1.183 Rev 0, depending on the resolutions of remarks pertaining to non LOCA gap fractions. RECOMMENDATION: Voluntary upgrade of AST analyses from conformance to RG 1.183 Rev 0 to RG 1.183 Rev 1 in the absence of any other issue requiring a LAR can be made without prior NRC review. Nonetheless, the staff should

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				assess whether a licensee given permission to fully implement the method of AST pursuant to RG 1.183 Rev 0 at his facility may conform to RG 1.183 Rev 1 without a license amendment request (LAR). If yes, the staff should state this clearly. If not, then considering the guidance in §C.1.2.1, the staff should give clear guidelines for submitting a LAR based on a voluntary revision of existing AST analyses from conformance to RG 1.183 Rev 0 to conformance to RG 1.183 Rev 1.
28	14	C.1.3.5	The first paragraph states, "***Current EQ analyses may be impacted by a proposed plant modification associated with AST implementation. The licensee should update EQ analyses that have assumptions or inputs affected by the plant modification to address these impacts.***"	QUESTION: Regarding the first paragraph; does this now mean that full EQ analyses are required when converting to AST? RECOMMENDATION: Provide more explicit guidance with respect to EQ doses.
29	14	C.1.3.5		COMMENTS: Applicants/licensees are expected to evaluate AST-related EQ analysis based on a Draft Guide DG-1239 that has not been issued. The full impact of DG-1199 and the unpublished DG-1239 cannot be determined until DG-1239 is publicly available. RECOMMENDATION: The comment due date for DG-1199 should be extended to allow time for an integrated review of both documents (i.e., after DG-1239 becomes publicly available).
30	14	C.1.3.5	The second paragraph states;; ***"For new facilities that are proposing to implement an AST and have EQ analyses impacted by a proposed plant modification***	QUESTION: Regarding the second paragraph; it discusses new reactors proposing to implement AST. Have any new reactors been licensed to TID-

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			associated with the AST implementation, the guidance that is being developed in a draft guide, Draft Regulatory Guide DG-1239, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plant," which will be published soon, should be used."	14844? <u>RECOMMENDATION:</u> Clarify the NRC's position regarding AST and new reactors.
31	14	C.1.4	The second sentence states; ***"These assumptions have no direct influence on the probability of the design basis initiator. These analysis assumptions cannot increase the core damage frequency (CDF) or the large early release frequency (LERF)."***	<u>COMMENT:</u> The Sandia MELCOR report now requires plants to analyze the LOCA with no ECCS injection for the first two hours. This will change the PRA analysis, impacting the large early release. Is a re-evaluation of the PRA analysis now required based on this new assumption? <u>RECOMMENDATION:</u> The Sandia MELCOR report appears flawed and should be re-evaluated to allow credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public comment period for DG-1199. Additional technical comments will be available for a future Regulatory Guide revision.
32	15	C.1.5	The first paragraph states; ***"The NRC staff accomplishes these reviews***	<u>COMMENT:</u> This statement is inconsistent with current

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			by evaluating the information submitted in the amendment request against the current plant design as documented in the FSAR, staff safety evaluation reports, regulatory guidance, other licensee commitments, and staff experience gained in approving similar requests for other plants."	practices regarding NRC reviews and approvals. What has recently been approved for one utility has not been allowed for another. This has been typical when different reviewers are involved, having different interpretations of the RG. This proposed revision does not ensure standardization among reviewers. RECOMMENDATION: Ensure standardization among reviewers.
33	16	C.2		COMMENT: This section provides five attributes to be considered for an acceptable AST, however this draft guidance is being presented as an 'acceptable AST' and those same attributes have not been met. For example, a defensible technical basis and empirical data (e.g., NUREG CR/XXXX, ANSI Standard 5.4,), independently verified and validated, and documented in a scrutable form has not been made available for public review and disclosure, and an independent peer review report, including resolution of comments, does not appear to exist. RECOMMENDATION: The draft guide presents a significant change to the previous AST, if not actually a "new" AST. It is unlikely that the NRC would permit a licensee to submit an application utilizing such a significant departure from a previously approved AST without meeting the listed attributes. The listed attributes should be met for the proposed changes in DG-1199 prior to finalizing the guide.
34	16	C.2 (a)	Bullet "a" states;	COMMENT:

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			"The AST must be based on major accidents hypothesized for the purposes of design analyses or consideration of possible accidental events that could result in hazards not exceeded by those from other accidents considered credible. The AST must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products."	This statement indicates that a substantial meltdown of the core must be considered with subsequent release of appreciable quantities of fission products. The word "appreciable" is not clear. This does not match up with what appears to have been done in the Sandia MELCOR report. RECOMMENDATION: Define how a licensee can determine how much of a release is "appreciable." The release used in the current revision appears appreciable. Therefore, a better description of "appreciable" is needed.
35	16	C.2 (c)	Bullet "c" states; ***"Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered."***	COMMENT (Editorial): Although this is unchanged from Revision 0, the NRC has been reluctant in the recent past to approve anything that has deviated from the RG. A statement in the body of Section 2 above states that the NRC staff does not expect to approve any source term that is not of the same level of quality as the source terms in NUREG-1465. By making this statement, licensees can expect that any deviation will not be approved. RECOMMENDATION: More consideration regarding deviations from this regulatory guide that "describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable in complying with alternative source term (AST) regulations for design basis accident (DBA) dose consequence analysis" must be afforded by the NRC reviewers.
36	16	C.2 (d)	Bullet "d" states;	COMMENT:

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			"The AST must have a defensible technical basis supported by sufficient experimental and empirical data, be verified and validated, and be documented in a scrutable form that facilitates public review and discourse."	It is not clear if the Sandia MELCOR report not deals with sufficient experimental and empirical data, or is verified and validated. The inputs used in MELCOR are not consistent with the way that plants are designed and operated. <u>RECOMMENDATION:</u> Perform the MELCOR runs using ESF systems in the way that they were designed. A specific run to develop the source term is not needed since one already exists (NUREG-1465). A new run using more accurate plant data (e.g., being able to run ECCS systems when they are designed (and tested) to function). Therefore, it is recommended that this analysis be undertaken, perhaps funded by the BWROG. The Sandia MELCOR report appears flawed and should be re-evaluated to allow credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public comment period for DG-1199. Additional technical comments will be available for a future Regulatory Guide revision.
37	18		The first paragraph states; ***"Table 1 (for BWRs) and Table 2 (for PWRs)***	<u>COMMENT:</u> This statement indicates that these tables are

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			list the core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs and non-LOCA DBAs where the fuel is melted and the cladding is breached."	to be used for both LOCA and non-LOCA events where fuel is melted. However, for RIA events, Table 4 is to be used. RECOMMENDATION: Clarify that Table 4 is to be used for non-LOCA RIA events.
38	18	3.2, 5th paragraph last sentence; and Appendix H		COMMENT: The NRC presents guidance for the radioactive source term for an AST analysis of the rod ejection accident (REA) in RG 1.183 Appendix H § 1. This guidance covers the potential for post REA fuel melt. In DG-1199 this guidance is stricken with the NRC stating that "The total fission product inventory for at-power RIA [REA] scenarios experiencing limited centerline fuel melt may be considered on a case-by-case basis (§ 3.2, 5th paragraph). This revision presents a problem for any licensee of a nuclear plant for which "limited centerline fuel melt" following an REA may be in the current license basis of that facility. It also creates the possibility of different source terms for REAs with "limited centerline fuel melt" based purely on different methodologies for developing that constituent to the source terms. RECOMMENDATION: The NRC staff should leave in RG 1.183 Appendix H some set of guidelines for developing a constituent to the REA source term associated with "limited centerline fuel melt." The guidelines should address the current discrepancies between Tables 1 & 2 & Footnote 11 and Tables 3 & 4 & Figure 1. If it

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				is the intent of the NRC to discourage the assumption of post REA fuel melt, they should make this clear in RG 1.183 Rev 1.
39				COMMENT: Tables 3 and 4 list revised fuel pin gap fractions for non LOCA events. Table 3 lists the baseline values for non LOCA events while Table 4 accounts for increases in gap fractions for the rod ejection (or rod drop) accident. Figure 1 defines the envelopes (in terms of peak nodal power and peak nodal burnup) over which these non LOCA gap fractions are valid. The basis documents for these proposed non LOCA gap fractions are Pacific Northwest National Laboratory (PNNL) Report PNNL-18212 (Ref. 2) and a proposed revision to the American Nuclear Society Standard ANS 5.4. The NRC staff has made PNNL-18212 available to the public through its ADAMS search engine. However, to date this report has not been reviewed within the nuclear industry. Also, to date, the ANS has not approved the revision to ANS 5.4. RECOMMENDATION: We propose that RG 1.183 Table 3 and Footnote 11 not be replaced with Tables 3 and 4 and Figure 1 of DG-1199 until the ANS approves the new version of ANS 5.4 and PNNL has reviewed it to determine if PNNL-18212 should be revised to account for any differences between the draft version used in it and the approved version. We also propose that the staff resolve comments received on

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				PNNL-18212 (attached at the end of these comments).
40			The third paragraph states; ***"Table 4 list the combined fission product inventory, by radionuclide groups, available for release for a fuel rod during a RIA. The transient fission product release component is presented as a function of increase in radial average fuel enthalpy (ΔH, cal/g)."***	COMMENT / QUESTION: Grammar edit: "Table 4 lists the…" It is understood that Table 4 is to be used for both PWRs and BWRs. Since they are not the same (PWR values too conservative for BWRs), there should be separate tables as there is for the LOCA (i.e., Tables 1 and 2). RECOMMENDATION: Correct grammar and create separate tables for BWRs and PWRs.
41	20	C.3.2		RECOMMENDATION: Figure 1 should be based on 10x10 fuel for BWRs since the majority of US BWRs operate 10x10 fuel.
42	18	3.2	The 4th paragraph says that Reference 18 (02/10/2009 NRC Memo) documents the methods used to calculate the Table 3 and Table 4 fission product inventories.	QUESTION: Which part of the memo describes the methods used to calculate the values in Table 3? Why is a memo being referenced instead of some type of regulatory document? RECOMMENDATION: Technical methods should not be documented in a memo. The source document for the memo needs to be evaluated. Therefore, additional review time is required after the document is made publicly available.
43	18	3.2		COMMENT: Based on the information provided by the NRC for the November 16, 2009 public workshop on DG 1199, there are significant differences between the calculated PWR and BWR Non-LOCA gap fractions.

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				RECOMMENDATION: Instead of identifying one limiting set of Non-LOCA gap fractions that are bounding for both PWR and BWR applications, Table 3 should list the Non-LOCA gap fractions for PWRs and BWRs separately. Also, Table 4 methodology should also use PWR and BWR Non-LOCA gap fractions.
44	18	3.2	The 5th paragraph says that Tables 3 and 4 are not applicable to fuel rods that experience fuel melting.	QUESTION: If the gap fraction is different due to the reactivity, why is the melt fraction during reactivity initiated events not also affected? If licensees are required to differentiate between BWRs and PWRs in Tables 1 and 2, then why are there not separate non-LOCA gap fraction tables for PWRs and BWRs as well? RECOMMENDATION: Create separate RIA tables for BWRs and PWRs. The source document for the memo needs to be evaluated. Therefore, additional review time is required after the document is made publicly available.
45	18	Section 3.2 (last para)	"The RIA combined release fractions provided in Table 3 and 4 of this guide are not applicable to fuel rods which experience fuel melting. ***The total fission product inventory for at-power RIA scenarios experiencing limited centerline fuel melting may be considered on a case-by-case basis***."	COMMENT Based on feedback provided by NRC at the 11/16/09 Workshop with the industry, in general, the PWR Control Rod Ejection accident and the BWR Control Rod Drop accidents are no longer expected to result in fuel melt. RECOMMENDATION For purposes of clarification, and to allow application of a conservative simplified approach in the event fuel melt is predicted, it

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				is suggested that the text be updated as follows: "The RIA combined release fractions provided in Table 3 and 4 of this guide are not applicable to fuel rods which experience fuel melting. **Reference X has determined that RIA scenarios rarely result in fuel melt. If fuel melt is postulated,** the total fission product inventory for at-power RIA scenarios experiencing limited centerline fuel melting may be considered on a case-by-case basis. **As a conservative alternative,** the fraction of fission products presented in **Tables 1 and 2 associated with the fraction of the fuel that is postulated to experience centerline melt following a RIA should be assumed to be released instantaneously to the containment or coolant as per accident scenarios defined for RIAs in Appendices in C and H."**
46	6	Section B (3rd para)	"The NRC staff considered the applicability of the revised source terms to operating reactors and ***determined that the current analytical approach based on the TID- 14844 source term would continue to be adequate to protect public health and safety***. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. However, the NRC staff determined that some operating reactor licensees might request to use an AST in analyses to support cost-beneficial licensing actions."	<u>COMMENT</u> The change in gap fractions being proposed by NRC reflects the results of new research and provides bounding values that support the removal of Note 11 to RG 1.183, Rev 0. This additional guidance is very useful to licensees. It is requested that NRC include the additional gap fraction data points that are available in the said research (for different fuel types, PWRs and BWRs, etc.) including the methodology utilized to determine the gap fractions in the finalized RG.
	18	C.3.2 (2nd para)	General Comment relative to new gap fractions	However, additional clarification needs to be included in Section B. With the RG 1.183, Rev

<table>
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<tr><td></td><td></td><td></td><td></td><td>0 gap fractions, the statement on Pg 6 (see quoted sentence in previous column) is valid as the gap fractions postulated in NUREG/CR 5009 for TID applications with extended burn fuel, <u>bound</u> those presented in RG 1.183 Rev 0.

However, with the new gap fractions presented in Table 3 of DG-1199, (and further increased as noted in Table 4 for the RIA events), the statement on pg 6 seems to be at risk. Provided below is a comparison of NUREG/CR 5009, RG 1.183 Rev 0, and DG 1199 gap fractions.

<table>
<tr><th>Group</th><th>NUREG/CR 5009 (TID)</th><th>RG 1.183 R0</th><th>DG-1199</th></tr>
<tr><td>I131</td><td>0.12</td><td>0.08</td><td>0.08</td></tr>
<tr><td>Kr85</td><td>0.3</td><td>0.1</td><td>0.35</td></tr>
<tr><td>I132</td><td>0.1</td><td>0.05</td><td>0.23</td></tr>
<tr><td>Other Hal</td><td>0.1</td><td>0.05</td><td>0.05</td></tr>
<tr><td>Other NG</td><td>0.1</td><td>0.05</td><td>0.04</td></tr>
<tr><td>Alkali</td><td>0.17</td><td>0.12</td><td>0.46</td></tr>
</table>

<u>RECOMMENDATION</u>
Based on the above it seems appropriate to either

a) reduce the conservatism included in the DG-1199 AST gap fractions or provide sufficient information to demonstrate that continued use of NUREG/CR 5009 gap fractions is acceptable for AST applications or

b) update the gap fraction values to be used by licensees with a non-AST licensing bases or update the text on pg 6 to include a basis why continued use of NUREG/CR 5009 gap fractions is acceptable for TID applications.</td></tr>
<tr><td>47</td><td></td><td>Tables 1 & 2</td><td></td><td><u>COMMENT:</u></td></tr>
</table>

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		versus Tables 3 & 4.		Currently, the non LOCA gap fractions (Table 3) for ^{85}Kr (10%), ^{131}I (8%), and the alkali metals (12%) are set to different values from the post LOCA gap phase release fractions for these isotopes (5%). These differences are made larger with the revisions to Table 3 and the new Table 4 proposed in DG-1199. Higher baseline non LOCA gap fractions are proposed for ^{85}Kr (35%) and the alkali metals (46%), significantly increasing the differences between them and the corresponding post LOCA gap release fractions. Also, revised baseline non LOCA gap fractions are proposed for ^{132}I (23%) and noble gases other than ^{85}Kr (4%), creating differences between them and the corresponding post LOCA gap phase release fractions. The baseline gap fractions (that is, not including the changes presumably induced with the REA) presumably indicate the fractions of various isotopes (noble gas, halogens, and alkali metals) in the core that are in the fuel pin gaps. As such, they correspond to limiting conditions of unit operation and are initial conditions for an event. They should take the same values for any event. In particular, they should take the same values as the gap phase release fractions for the LOCA. The proposed value to the non LOCA gap fraction for alkali metals is a case in extreme. The proposed value of 46% is higher than the release fraction for the post LOCA gap and early in-vessel phases combined (30% for

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				pressurized water reactors - PWRs - and 25% for boiling water reactors - BWRs). This discrepancy does not square with the presumption of the radioactive source term for the LOCA corresponding to damage to the fuel pellets as well as the cladding of the fuel pins. <u>RECOMMENDATION:</u> The staff should address the apparent discrepancy between the proposed non LOCA gap fractions and the post LOCA gap release phase fractions.
48	18	Footnote 11 versus Figure 1 (compare with RG 1.183 Footnotes 10 & 11).		<u>COMMENT:</u> Footnotes 10 and 11 of RG 1.183 define the envelope over which the guidelines in § 3.2 apply and in particular over which the release fractions of Tables 1-3 are valid. In DG-1199 the NRC proposes to replace Footnotes 10 and 11 with Footnote 11 and Figure 1. Presumably, in DG-1199 Footnote 11 defines the envelope of validity of the post LOCA release fractions of Tables 1 and 2; Figure 1 defines the envelopes of validity for the non LOCA gap fractions in Tables 3 and 4. As defined, the envelope of applicability for non LOCA gap fractions (Figure 1) includes a variable range "nodal" power while the envelope of applicability for post LOCA release fractions (Footnote 11) includes no such range. The burnup range in the envelopes of validity for the post LOCA release fractions and non LOCA gap fractions now are different. The post LOCA gap phase release fractions

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				and the non LOCA gap fractions are initial conditions for an accident and as such are not affected by the accident. Accordingly, it is reasonable to expect that these fractions would be valid over the same envelope of peak nodal power and burnup (peak rod average, peak pellet, or peak nodal). <u>RECOMMENDATION:</u> Consider defining one envelope for each reactor type (BWR and PWR) over which all release fractions are valid. At a minimum the staff should consider defining one envelope for each reactor type over which the post LOCA gap phase release fractions and the non LOCA gap fractions are valid.
49	19	Tables 3 & 4 & Figure 1 versus RG 1.183 Table 3 & Footnote 11).		<u>COMMENT:</u> Evidently, the non LOCA gap fractions are being revised to accommodate unit operations with fuel power and burnup beyond the envelope defined by RG 1.183 Footnote 11. This is made evident in an internal NRC memorandum (Ref. 1) and Report PNNL-18212 (Ref. 2). Some licensees still design and operate nuclear fuel the performance characteristics of which still lie in the envelope of RG 1.183 Footnote 11 (as well as Footnote 10 pertaining to post LOCA release fractions). The values in RG 1.183 Table 3 for non LOCA accidents should still apply to fuel designed and operated in conformance to the current limits. <u>RECOMMENDATION:</u> When revising RG 1.183, the NRC staff should consider retaining the current Table 3 for that

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				fuel which conforms to the envelope defined in RG 1.183 Footnote 11. As another alternative to resolve this remark, the staff could consider replacing DG-1199 Tables 3 and 4 and Figure 11 with a methodology by which licensees could calculate non LOCA gap fractions for individual nodes or sets of nodes in a reactor core. The methodology would allow the calculation of separate non LOCA gap fractions and post RIA increases in gap fractions based on the reactor type, fuel type, burnup and LHGR. This methodology could take the following forms: 1) The staff endorses a set of correlations by which separate values could be calculated for non LOCA gap fractions and post RIA increases to the gap fractions. The correlations could allow the calculation of different gap fractions based on reactor type, fuel type, burnup and LHGR (within the envelope of Figure 1). This option is preferred. 2) The NRC staff endorses a set of several tables to replace Table 3 for the listing of separate values for non LOCA gap fractions. The NRC staff also endorses a set of several tables to replace Table 4 for the listing of separate values for increases to gap fractions following a RIA. The tables would list separate values based on reactor type, fuel type, burnup and LHGR.

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				3) The staff endorses the version of the computer code FRAPCON used in the study supporting the new non LOCA gap fractions and reported in PNNL-18212. The staff also provides guidelines for the use of this version of FRAPCON to calculate separate values for both the non LOCA gap fractions and RIA induced increases to them based on reactor type, fuel type, burnup and LHGR. The staff makes available or causes to be made available both this FRAPCON version as well as a user manual for it. This option is least preferred.
50	20	Figure 1.		<u>COMMENT:</u> The envelopes in Figure 1 are defined in terms of peak nodal power and peak nodal burnup. Nowhere in DG-1199 is the term "nodal" defined. This ambiguity may give rise to uncertainty among some licensees as to whether their fuel is operated within the envelopes. <u>RECOMMENDATION:</u> The NRC staff should define the term "nodal." Consider defining "nodal" as was used by PNNL in its work (Ref. 2).
51	20	Figure 1.		<u>COMMENT:</u> The boundary of the envelopes in Figure 1 may be beyond the nodal power and burnups in reactor cores in at least some plants. <u>RECOMMENDATION:</u> We propose that the staff replace Tables 3 and 4 and Figure 1 with a methodology by which licensees would calculate limiting gap fractions

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				for AST analyses of non LOCA events at their plants. Refer to the resolution proposed for Item 53 for additional details.
52	6 18	B (3rd para) C.3.2 (3rd para)	"Facility-analyzed DBAs ***are not intended to be actual event sequences; rather, they are intended to be surrogates*** to enable deterministic evaluation of the response of engineered safety features (ESFs)." General Comment relative to development of releases from a damaged fuel rod after a RIA	COMMENT Relative to RIA, the development of the increase in radial average fuel enthalpy (cal/g) in different axial regions will require 3D modeling (which may be impacted with each re-load) and will require significant analytical effort on part of the licensee. RECOMMENDATION Since (see quoted sentence in previous column), these DBA's are intended to be conservative surrogates of the real event, NRC should include in the final RG (based on the research already conducted), a conservative alternative to plant specific (and perhaps re-load specific) enthalpy increases calculations, such as: a) bounding peak enthalpy increases for a simple 3 region model that could be used as conservative "default" values or b) a suitable multiplier to the gap fractions presented in Table 3 to develop releases due to a RIA
53	18	C.3.2	Section text and Figure 1	RECOMMENDATION: Define "axial node" as it is used in this section of the guide.
54	19	3.2	NRC Presentation on non-LOCA Gap Fractions at the DG-1199 Workshop on 11/16/09	COMMENT: PWR fuel vendors apparently do not currently have 3D enthalpy rise limits for RIA events (ΔH, cal/g). The example of the 3D RIA Gap

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				Fraction Calculation for PWR fuel provided in the workshop handouts, the Gap Fraction Basis memo (ML090360256), and PNNL-18212 implies that 3D enthalpy rise limits could be applied based on axial region to determine a weighted gap release by isotope. Without 3D enthalpy rise limits calculated gap releases for the RIA event are prohibitively large. <u>RECOMMENDATION</u>: Please include the 3D RIA Gap Fraction Calculation in the final guide and indicate whether or not the peak enthalpy rise values used in the example are generically applicable to PWR fuel or need to be determined on a plant specific basis. This also applies to some BWR fuel vendors that do not use 3D enthalpy modeling.
55	19	C Section 3.2 Tables 1, 2, and 3	Tables 1 and 2 contain no changes in release fractions.	<u>QUESTION:</u> Table 3 contains a large increase in the alkali metals fraction. It is not clear how there can be such a large increase in the Table 3 value. Also, Table 3 needs to be different for BWRs and PWRs. <u>RECOMMENDATION:</u> Table 3 needs to be different for BWRs and PWRs. The source document for the memo needs to be evaluated. Therefore, additional review time is required after the document is made publicly available.
56	19	C.3.2 Table 3	NRC Presentation on non-LOCA Gap Fractions at the DG-1199 Workshop on 11/16/09	<u>COMMENT</u> The NRC presentation on the development of the non-LOCA gap fractions presented some important assumptions that likely should be

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				considered if a licensee needs to evaluate new non-LOCA gap fractions. The assumptions include: (i) operation at 90% of the LHGR limit for nearly the entire rod life, (ii) operation on the LHGR limit for some period of time (~6 months), and (iii) application of a bounding 95/95 approach to address uncertainties. <u>RECOMMENDATION</u> These acceptable assumptions should be reported somewhere like a Reg Guide appendix or a separate technical document.
57	19	C.3.2 Table 3		<u>COMMENT</u>: The specific basis and methodology for increased gap fractions listed in Table 3 (compared to the values in RG 1.183 Rev 0) is unclear. This includes, for example, I-132, Kr-85, and alkali metals. <u>RECOMMENDATION</u>: Clarify or provide reference to the specific basis and methodology for increased gap fractions listed in Table 3. Also, please clarify which types of accidents these values are applicable to (e.g., all or fuel damage only).
58	19	C.3.2 Table 3		<u>RECOMMENDATION:</u> Please clarify the impact of DG-1199 (when finalized) on the ISG-5 source term and accident analysis. For example, will ISG-5 be updated and/or is it still applicable?
59	19	Section 3.2 Table 3	NRC Presentation on non-LOCA Gap Fractions at the DG-1199 Workshop on 11/16/09	<u>COMMENT</u>: The NRC presentation on the development of

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				the non-LOCA gap fractions indicated that the analysis applied a bounding 95/95 approach to address uncertainties. Section 3.6 of NUREG-1465 describes the LOCA release fractions as "representative or typical, rather than conservative or bounding values". The gap fractions currently reported in Reg Guide 1.183 are also best-estimate values. RECOMMENDATION: Consistent with the previous approaches, the proposed non-LOCA gap fractions in DG-1199 should therefore also be best-estimate values without the application of the 95^{th} percentile adders.
60	19	C.3.2 Table 3		COMMENT: The non-LOCA gap fractions proposed in DG-1199 are based on the Staff letter dated February 10, 2009. This letter indicates that the BWR gap fractions are significantly lower than the PWR values. RECOMMENDATION: A separate set of gap fraction tables should be reported for use by BWRs.
61	20	C.3.2 Table 4	The Table 4 gap fractions for RIA events causes extraordinarily high gap fractions (i.e., >1) for BWRs that do not calculate delta-h in three separate sections.	QUESTION: The alkali metals fraction will be 1.08 for a delta-h value of 200 cal/gm (1.17 for 230 Cal/gm) for a BWR that does not calculate delta-h in three separate sections. RECOMMENDATION: Determine a maximum fraction of less than 1 that should be used.

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62	20	C.3.2 Table 4		RECOMMENDATION: Please provide a reference for the technical basis for Table 4.
63	20	C.3.2 Table 4	Footnote #12 states; ***"ΔH = increase in radial average fuel enthalpy, cal/g"***	COMMENT: This footnote makes mention to "*radial average fuel enthalpy*". There should also be a footnote that discusses the axial fuel enthalpy as used in the examples in the 2/10/09 NRC Memo. RECOMMENDATION: The example given in Reference 18 should be included in the RG and annotated for better clarity. The source document for the memo needs to be evaluated. Therefore, additional review time is required after the document is made publicly available.
64	20	C.3.2 Figure 1		COMMENT: The fuel pellet power history envelope provided in Figure 1 (and footnote 11 on page 18) is only applicable to 70 GWD/MTU pellet exposure for BWRs. Some plants use AREVA ATRIUM-10 fuel which has been licensed to a fuel rod average exposure limit. The proposed power history envelope would restrict fuel exposure well below the ATRIUM-10 fuel licensed limit and would negatively impact fuel cycle management since a larger reload batch fraction would be necessary to remain below the proposed 70 GWD/MTU pellet exposure limit. RECOMMENDATION: To address this issue, we recommend one of the following approaches: (1) provide an option to implement the existing RG 1.183 power

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				history curve, allow extrapolation of the proposed power history curve to 80 GWD/MTU; (2) specify the limit in terms of average fuel rod exposure (consistent with the AREVA fuel licensing basis and similar to the current RG 1.183 basis); or (3) provide an acceptable method or procedure for a licensee to develop a custom power history curve which can be submitted in their own AST application.
65	20	C.3.2 Figure 1	The figure makes reference to peak nodal power and nodal burnup.	QUESTION: What is meant by a "node"? RECOMMENDATION: There needs to be a detailed description as to what is being referred to as a node.
66	20	C.3.2 Figure 1		RECOMMENDATION: Please define the terms peak nodal power and nodal burnup with regards to Figure 1 and indicate whether there are limitations on node size.
67	20	C.3.2 Figure 1	Figure 1 reports the maximum allowable power operating envelope for non-LOCA gap fractions	COMMENT: This figure refers to "nodal" power and burnup, which typically is based on how the rod is divided up into axial "nodes". For BWRs, rods are typically analyzed with 25 axial segments of 6-inch nodes. RECOMMENDATION: Describe the meaning of "nodal" in terms of this figure.
68	20	C.3.2 Figure 1		COMMENT: The power versus exposure curves presented in the Draft Reg Guide 1.183 Revision 1 do not envelope reactor operations currently supported by some vendors.

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				RECOMMENDATION: Since the NRC/PNNL claim the release fractions presented are conservative, the curves should be removed and only a footnote should be included within the guide indicating that the presented gap release fractions are bounding to peak fuel rod average exposures up to 62 GWd/MTU. An important point to make is that the Reg Guide should be written in a way as to NOT preclude the possibility for fuel rod average exposures beyond 62 GWd/MTU in the future. A path and/or acceptable method for calculating approved source terms at higher burnup should be included in the Reg Guide. (i.e. the AST Reg Guide should not dictate the industry's maximum allowable fuel exposure)
69	21	C.3.4	The new Table 6 contains a new radionuclide group (Barium, Strontium). The old Table 5 did not.	QUESTION: Barium and strontium were previously part of the Tellurium group. What is the technical significance of this change? Is RADTRAD affected? RECOMMENDATION: Provide a technical explanation as to why this is being done.
70	22	C.3.6	The 2nd paragraph says that ***"For the postulated main steamline break, steam generator tube rupture, and locked rotor accidents, the licensee should evaluate the amount of fuel damage assuming that the highest worth control rod is stuck at its fully withdrawn position."***	QUESTION: RG 1.183, Revision 0 requirements are: BWR MSLB: not a requirement. PWR MSLB: only required if fuel damage postulated. PWR SGTR: not a requirement PWR LRA: not a requirement Since most MSLB and SGTR accidents do not postulate fuel damage, additional analysis will

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				be required since this is not currently an assumption (except for the PWR MSLB). Does this mean that we must now consider the potential for fuel damage CAUSED due to the highest worth control rod being stuck fully withdrawn? Will this now make all these accidents into RIA events? <u>RECOMMENDATION:</u> Provide clarification of the NRC's intent regarding this change as well as any technical justification for this change.
71	22	C.3.6	***"For the postulated main steamline break, steam generator tube rupture, and locked rotor accidents, the licensee should evaluate the amount of fuel damage assuming that the highest worth control rod is stuck at its fully withdrawn position."***	<u>COMMENT</u>: This appears to be a single active failure. <u>RECOMMENDATION</u>: NRC should clarify if this is the NRC-required single active failure associated with these accidents or whether the licensee should select the worst failure for the specific plant.
72		Page 23, § C.4.1.5, 2nd paragraph & last sentence; and Page 23, § C.4.1.5, 3rd paragraph.	The last sentence of the 2nd paragraph in DG-1199 states that "In calculations, the maximum 2-hour EAB χ/Q [atmospheric dispersion factor] should be used for the entire duration of the release to the environment.... The 4th paragraph states "For the duration of the event, the breathing rate ... should be assumed to be 3.5×10^{-4} cubic meters per second." The staff has written these guidelines "to ensure that the limiting case [2 hour interval of maximum releases] is identified."	<u>COMMENT:</u> As written, the first two passages could be misinterpreted as guidelines to take both a constant *χ*/Q and breathing rate for the thirty day LPZ radiation dose. <u>RECOMMENDATION:</u> Replace "In calculations" with "In this calculation" or "In this effort" in the last sentence of the 2nd paragraph. Revise the 4th paragraph so that it states "For this effort, the breathing rate should be set to 3.5×10^{-4} cubic meters per second over the assumed duration of the event."
73	26	C.4.3	The reference to integrated radiation exposure of plant equipment has been removed from this section.	<u>COMMENT:</u> The reference to integrated radiation exposure

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				of plant equipment has been removed from this section. <u>RECOMMENDATION:</u> Provide information regarding where this is now located within DG-1199.
74	28	C.5.1.1	The second paragraph states; ***"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."***	<u>COMMENT:</u> It appears that the staff may have selected inappropriate assumption and models for use in the Sandia MELCOR report regarding ECCS injection times. <u>RECOMMENDATION:</u> Perform the MELCOR runs using ESF systems in the way that they were designed and approved for use. A specific run to develop the source term is not needed since a non-mechanistic source term already exists (NUREG-1465). A new run using more accurate plant data (e.g., being able to run ECCS systems when they are designed and tested to function) needs to be performed. The Sandia MELCOR report appears flawed and should be re-evaluated to allow credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public

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				comment period for DG-1199. Additional technical comments will be available for a future Regulatory Guide revision.
75	28	C.5.1.1	The second paragraph states: "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."	COMMENT: At what point do the conservative assumptions become excessive? A crtiterion should be provided for determining what is "sufficiently conservative." RECOMMENDATION: Provide a definition or measure that can be applied to data and analysis, for example, that data or analysis follow good engineering practice, while accounting for a reasonable numer of failures, or using the mean plus one standard deviation.
76	28	C.5.1.2	This section now states; "***However, the licensee should not take credit for engineered safeguards features that would affect the generation of the source term described in Tables 1 and 2. For example, licensees should not credit emergency core cooling system operation during the first two hours of the DBA in order to reduce or mitigate the source term generation in the core.***"	COMMENT: It is understood that the initial source term used in AST (the phased release from the core) is from Tables 1 and 2 and cannot be altered. It is further understood that ECCS operation such as sprays can be assumed. This will not affect the generation of the source term, but will affect the mitigation once it exits the core. Is this still correct? RECOMMENDATION: Perform the MELCOR runs using ESF systems in the way that they were designed. A specific run to develop the source term is not needed since one already exists (NUREG-1465). A new run using more accurate plant data (e.g., being able to run ECCS systems when they are designed and tested to function). The Sandia MELCOR report appears flawed and should be re-evaluated to allow

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				credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public comment period for DG-1199. Additional technical comments will be available for a future Regulatory Guide revision.
77		Page 30, § C.5.3, 2nd paragraph, 5th-10th sentences (that is, the last 6 sentences).		<u>COMMENT:</u> The NRC staff evidently is providing here the guidelines in RG 1.194 § 2 for the control room χ/Qs and in particular the assignment of the 0-2 hour value of the control room to the 2 hour interval of "limiting portion of the release to the environment." The staff also is providing in this passage guidelines for the calculation of the LPZ χ/Qs. The construction of this passage is not clear and allows the misinterpretation that a 0-2 hour LPZ χ/Q is to be calculated and assigned to the 2 hour interval of maximum release to the environment. This evidently is not the intention of the staff given the statement in the 3rd sentence that "the LPZ X/Q value [is generally determined] for a 0-8 hour averaging period." <u>RECOMMENDATION:</u> The staff can preclude any confusion over §

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				5.3 by replacing the last 7 sentences by a statement equivalent to the following: The staff endorses the guidelines of RG 1.194 § 2 to be an acceptable methodology for calculating the control room X/Q values." If the staff desires to explicitly restate the guidelines for RG 1.194 § 2 here, we recommend that they do the following: 1) Strike the 5th sentence ("The period of time of the most adverse...."). It already is redundant to the 7th sentence ("The 0-2 X/Q value should be ...."). 2) Strike the phrase "and LPZ" from the 6th sentence
78		Page 30, § C.5.3, 2nd paragraph, 6th-10th sentences (that is, the last 5 sentences).	[Note – This comment cross-references to "Note 1"]	COMMENT: In writing DG-1199 § 5.3 the staff has indicated that the 0-2 hour value for the control room χ/Q should be taken for the 2 hour time span of maximum releases to the environment over the duration of the presumed release. The guidelines in the last five sentences of the 2nd paragraph of this section conflicts with this expectation since it appears to limit the search for the maximum release to the first eight hours of the event. That is, the 8th sentence states that "The 2-8 X/Q value is used for the remaining 6 hours of the first 8-hour time period." The staff may mean to expand this guidance in writing the last sentence "The 8-24, 24-96, and 96-720 hour X/Q values should be similarly used for the remainder of the release duration." However, that is not clear and it could be taken to indicate that the 8-24,

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				24-96, and 96-720 hour control room χ/Q values should be used for the remaining intervals. Furthermore, no guidance is given for the case in which the 2 hour interval of maximum releases overlaps two abutting averaging intervals (say 0-8 and 8-24). <u>RECOMMENDATION:</u> In proposing this resolution we assume that the guidelines mandate the application of the 0-2 hour control room χ/Q value in the 2 hour interval of maximum release over the presumed duration of release (see DG-1199 Table 20). The staff should rewrite the 2nd paragraph to provide clear guidelines for assigning the 0-2 control room χ/Q value to any 2 hour time span in the presumed release duration. The guidelines should account for the possibility that this 2 hour time span may overlap two abutting averaging intervals. For staff consideration, a draft resolution of this remark is provided here as follows: Replace the last two sentences of the 2nd paragraph of § C.5.3 with the following or equivalent (at the end of the paragraph): If the limiting 2 hour interval falls within the within the averaging period 0-8 hours, the 2-8 hour control room χ/Q value is applied to the remaining 6 hours of this interval. If the limiting 2 hour interval falls within any other averaging period, the control room χ/Q value for the remainder of the period may be obtained using an averaging formula. If the limiting 2 hour interval overlaps two adjacent averaging period, separate averaging formulae may be applied to obtain a control room χ/Q value for each of the two averaging periods. Separate formula are used to account

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				for the portion of the limiting 2 hour interval in each of the abutting averaging periods. An alternative draft resolution listing formulae which may be applied in placing the 0-2 hour control room χ/Q value in the 2 hour time span of maximum releases to the environment is presented in Note 1 following all remarks on DG-1199 and PNNL-18212.
79	33	Regulatory Analysis, Alternative Approaches	Alternative #2 states: "...***maintain public safety by ensuring that safety analyses use appropriate analysis assumptions and methods***..."	COMMENT: This only maintains public safety for future analyses, not current analyses. Although the Backfit Analysis indicates that it does NOT require a backfit, the statement in item 1 appears that way. How can this statement be made without requiring a backfit? RECOMMENDATION: Re-evaluate this statement.
80	33	Regulatory Analysis		COMMENT: The impacts analysis in the DG only reports the minimal Staff resources to issue the guide as well as the cost savings associated with more efficient review of new reactor applicants and existing reactor licensee submittals. It further indicates that licensees would realize savings via the efficiencies gained in minimizing follow-up questions and revisions associated with each licensee application or amendment submittal. There would likely be considerable more cost associated with complying with DG-1199 than stated. Specifically, the proposed MSIV model changes in Section A-5 will significantly increase the doses from the MSIV release pathway. These increases will likely result in

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				more than a minimal increase in the consequences of an accident and thereby cannot be implemented via 10CFR50.59. Consequently, these changes would require licensee submittals to the NRC and Staff reviewer resources. In some cases, plant design changes may also be necessary to mitigate the dose increase or merely to meet the regulatory acceptance criteria in 10CFR50.67. These costs are considerably more than those discussed in the regulatory analysis. <u>RECOMMENDATION</u>: The regulatory analysis for the guide should be revised to better reflect impacts and increased regulatory burden that are reasonably expected to occur as a result of the significant changes that are being proposed in the guide.
81	33	Regulatory Analysis, Alternative Approaches	Alternative #2 states: "...***reduce unnecessary regulatory burden by providing clear AST methods and assumptions for dose consequence analysis***..."	<u>COMMENT:</u> It is not clear how this will reduce the regulatory burden. The burden for industry has just been greatly increased since new analyses will now be required with respect to new, more complicated gap fractions and steam line deposition models. This means all the more documents for the NRC to review with AST submittals. Not only is the information in the DG unclear, it is also very complicated in that licensees will now need to refer to several documents since the new guide does not stand alone. Additionally, it is likely that the BWROG may submit a Topical Report AFTER the RG revision is finalized in accordance with the current public comment schedule. Therefore, the current schedule will

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				increase the burden for both industry submittal preparation and regulatory review. <u>RECOMMENDATION:</u> Re-evaluate this statement.
82	33	Regulatory Analysis, Alternative Approaches	Alternative #2 states: "***improve efficiency and effectiveness , as the revised guidance would provide licensees with the staff positions, thereby minimizing RAIs and resubmittals,***"	<u>COMMENT:</u> The new guidance is very complicated, not clear, and will require additional analyses to be submitted, most likely leading to even more RAI questions. This will also provide at least TWO methods for AST analysis that the NRC must contend with (Rev. 0 and Rev. 1). <u>RECOMMENDATION:</u> Re-evaluate this statement.
83	34	Regulatory Analysis, Evaluation of Values and Impacts	The top bullet on this page states; ***"Regulatory Guide 1.183 has improved regulatory efficiency by providing an acceptable approach and by encouraging consistency in the assessment of control room habitability and offsite accident consequences."***	<u>COMMENT:</u> There will most likely not be much consistency once this RG is approved. It is unlikely that BWRs would commit to using it due to extreme penalties with regard to MSIV leakages. <u>RECOMMENDATION:</u> Re-evaluate this statement.
84	34	Regulatory Analysis, Evaluation of Values and Impacts	The top bullet on this page states; ***"The revised guide would reduce the likelihood for followup questions and possible revisions in licensees' analyses and plant modifications. The proposed regulatory guide would simplify NRC reviews because license applications and amendments should be more predictable and analytically consistent."***	<u>COMMENT:</u> This revision is very complicated, lending itself to more opportunities for followup questions. <u>RECOMMENDATION:</u> Re-evaluate this statement.
85	34	Regulatory Analysis, Evaluation of Values and	The second bullet on this page states; ***"The revised regulatory guide would result in cost savings to both the NRC and industry."***	<u>COMMENT:</u> This revision would require additional cost for licensees due to the fact that additional analyses and plant modifications would be

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		Impacts		required in order to meet the MSIV leakage doses using this new guidance. <u>RECOMMENDATION:</u> Re-evaluate this statement.
86	34	Regulatory Analysis, Backfit Analysis	The first paragraph states; ***"The proposed regulatory guide revision does not require a backfit analysis as described in 10 CFR 50.109(c) because it does not impose a new or amended provision in the NRC's regulations. It does not impose a regulatory staff position that interprets the NRC's regulations differently than a previously applicable staff position."***	<u>COMMENT:</u> The previously applicable staff position was based on the premise that the MSIV leakage source term was the same as that in the drywell. The new position is that, not only is there a new source term to consider, there is also a new method to determine what that source term should be (i.e., MELCOR). Granted, no "regulations" are affected, but the NRC-accepted method to meet the regulations is greatly affected. For plants already licensed for AST, what is the affect if the licensee desires to make a new submittal to change a dose that has nothing to do with MSIV leakage? Will plants be required to use the new revision for ALL future submittals, no matter what the reason? If Revision 0 is in a plant's licensing basis, can it still use Revision 0 for changes using 10 CFR 50.59? Since the paragraph says licensees can select a preferred method of achieving compliance with a license condition, can we still use Revision 0 since that is our commitment in the UFSAR? <u>RECOMMENDATION:</u> Provide additional guidance, perhaps using examples, with respect to these questions.
87	35	Regulatory Analysis, Backfit Analysis	This section states; ***"The NRC staff will use this guide to evaluate licensee-initiated changes if there is a clear nexus between the proposed change and the***	<u>COMMENT:</u> These statements are not clear with respect to when a plant already licensed to RG 1.183, Rev. 0.

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			guidance contained in the guide. The staff will also use it to review changes when the licensees have committed to using this guide."	RECOMMENDATION: Clarify the NRC's intent with regard to these statements.
88	36	Reference 1		QUESTION: Is there any intent to revise NUREG-1465 to account for newer release fractions and timings similar to those in the Sandia report referenced in Appendix A? Should this draft guide be held for the release of the new source term and an update to the companion SECY 98-154 (Reference 14) rebaselining document?
89		Appendix A, § A-2.2		COMMENT: The NRC staff endorses credit for natural deposition of fission product onto containment internal structures following a LOCA with the conditions stated in this section. These stated conditions do not include any concerning credit for containment spray. The NRC staff evidently has objected to credit for natural deposition in the sprayed region of containment (see W.R. McCollum - Duke Energy - to USNRC dated May 20, 2002, Attachment 1). RECOMMENDATION: The staff should review this section in light of correspondence such as the one cited here and amend this section to state any limitations on credit for natural deposition in containment regions in which credit for containment spray is taken.
90		Appendix A § A4.4 & A4.5		COMMENT: Per the guidelines in § A4.4, the partition fraction for iodine airborne from Engineered

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				Safety Features (ESF) leakage in the auxiliary building is to be set equal to the leakage flash fraction only. Per the guidelines in § A 4.5, the iodine partition fraction is to be set to a lower bound of 10% "unless a smaller amount can be justified." These guidelines supposedly account for evaporative partitioning of iodine from the pool of ESF leakage to the air. Taken together, these guidelines are not consistent for the following reasons. First they allow evaporative iodine partitioning from ESF leakage to be neglected with a flash fraction above 10%. Second, they promote a smaller iodine partition fraction to be associated with evaporative transfer for scenarios with flashing than the value this constituent would take with less or no flashing. Conversely, the guidelines may promote an inordinately large value for a lower bound to the iodine partition fraction. <u>RECOMMENDATION:</u> The staff should consider rewriting the guidelines in § A 4.4 and A4.5 to promote setting the iodine partition fraction for post LOCA ESF leakage in the Auxiliary Building to the sum of the flash fraction and 10% of the leakage that does not flash. Also, the staff may want to consider a smaller value for the lower bound for the iodine partition fraction for post LOCA ESF leakage in the auxiliary building.
91		Appendix A § A4.5		<u>COMMENT:</u> Currently, the staff is reviewing the STARFIRE methodology for the calculation of the partition fraction for iodine airborne from ESF leakage. The outcome of the review may dictate

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				changes to the guidelines in this section. <u>RECOMMENDATION:</u> The staff may want to consider suspending any revision to the guidelines in this section until it completes its review of the STARFIRE methodology.
92	A-1	Appendix A (intro)	"***As such, the licensee should analyze the spectrum of large-break LOCAs credible for its facility. The analysis should determine the limiting large-break LOCA, assuming substantial core damage, from the perspective of dose consequences to the public and control room workers.***"	<u>COMMENT:</u> Does this mean that we no longer need to assume a steam line break in BWR containments as well as the large-break LOCA (largest size pipe in the RCS)? This "requirement" came about when licensees started to credit main steam piping between the RPV nozzle and the inboard MSIV for iodine deposition. Some Accident Dose Assessment Branch reviewers did not like the fact that all four inboard steam lines were being credited. Since BWRs already have inboard MSLB accidents evaluated to show that fuel damage does not occur, why must we assume TWO piping failures? <u>RECOMMENDATION:</u> Clarify the NRC's intent with respect to these statements.
93	A-6 6	A-5.1 Section B (3rd para)	General comment relative to development of the MSIV leakage source term "The NRC staff considered the applicability of the revised source terms to operating reactors and ***determined that the current analytical approach based on the TID- 14844 source term would continue to be adequate to protect public health and safety***. Operating reactors licensed under that approach would not be required to re-analyze	<u>COMMENT</u> The change in treatment of MSIV leakage being proposed by NRC needs further consideration. Traditionally, radiological analyses have always been based *on "a conservative source term or radioactivity release into the containment"* (that postulates a fuel melt based on delayed ECCS), *and radioactivity transport*

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			accidents using the revised source terms. However, the NRC staff determined that some operating reactor licensees might request to use an AST in analyses to support cost-beneficial licensing actions."	*that takes into consideration plant design* (which includes safety related equipment, redundancy, seismic design, etc). The very essence of this approach is being challenged with the new MSIV leakage model relative to increased activity in the steam dome. Note that whether a plant used a TID 14844 source term or an AST source term, the fuel melt postulated with either approach *represents a significant delay in ECCS initiation*. The change in the treatment of MSIV leakage proposed by the DG is not a source term issue (since both source terms result from delayed ECCS) - it's a transport issue. It appears that relative to transport, the TID model is currently not required to address a delay in reflood, whereas the AST model will be required to address a 2hr delay with a consequent fission product holdup in the reactor vessel. Currently, both TID and AST models acknowledge that fuel damage / melt occurs after initial blowdown, and that the fission products in the vessel are released to the containment /drywell during reflood. If credit is taken for the operation of ECCS as it is designed, reflood will occur right away, and the conservative postulated accident source terms / releases, whether TID or AST, will be discharged into the drywell as it is released from the core. *Based on an implicit assumption of ECCS operation as designed, the current assumption of the steam dome and the drywell being at the same concentration is a*

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				reasonable surrogate event for both TID and AST models. It is understood that delayed ECCS initiation due to human errors / lack of strict rules regarding maintenance practices, etc. was the cause of the fuel damage experienced at TMI. These concerns were addressed under TMI Action items and the changes imposed by NRC were implemented by all plants. If NRC believes that the above actions are insufficient and that as an added conservatism, *activity transport should also reflect the postulated delay in ECCS*, there appears to be no reason why this change is only applicable to AST applications. By taking the approach proposed in the DG, BWRs with AST will be forced to tighten up the MSIV leakage pathway to reduce dose consequences or expend significant effort/cost to defend a seismically rugged condenser system or to re-activate leak collections systems that have high maintenance costs. This type of change in approach should be investigated carefully as it could have far-reaching and as noted above, cost-intensive impacts which are not necessarily safety significant. For e.g., the DG implies that the need to address an increased concentration in the steam dome is a natural progression from the basis of the alternative source term, (i.e., a two hr delay in ECCS), thus this delay needs to be

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				addressed in the MSIV leakage transport model. The obvious downstream question is how far should this "consistent transport model" be taken? For e.g., does the activity burst into drywell/containment at the end of the 2 hr period as a result of reflood need to be addressed / evaluated for continued availability of credited safeguards systems and for dose impact? Does the impact of this delay in release, on the currently used aerosol characteristics in the drywell /containment, need to be addressed? These types of issues are applicable to both PWRs and BWRs. Note that traditionally, a "conservative" source term was used to evaluate the effectiveness of the "as designed" containment and plant safety systems to control the radiological consequences of an accident. With the inclusion of ECCS failure in the transport model, the definition of the design basis surrogate LOCA for dose consequences has now changed to explicitly include ECCS failure for 2 hrs. A logical progression might require the industry to address a recovery plan (within 2 hrs) for core cooling. Note that there seems to be no clear reason why an ECCS system failure needs to be specifically considered, given that all of the safety related systems have the same design basis requirements relative to quality and redundancy. In summary, it is concluded that the change in treatment of MSIV leakage, and inclusion of

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				ECCS failure as an explicit part of the DBA, needs further consideration. <u>RECOMMENDATION</u> It is suggested that the RG should either: a) Include text that addresses the concerns raised and the validity of the statement made on pg 6 (see quoted sentence in previous column), or Revert back to the AST model postulated in RG1.183, Rev 0
94	A-6	Section A-5.1	"The source of the MSIV leakage is assumed to be activity concentration in the reactor vessel steam dome. At the end of the early in-vessel release phase, the activity concentration in the vessel dome should be assumed to equal the containment (or drywell activity concentration".	<u>COMMENT:</u> These statements imply the need for a new and unprecedented consideration of a <u>primary containment bypass pathway</u> in addition to the conventional secondary containment bypass path assessment. <u>Containment</u> response would now be based on the assumptions used to establish core source terms, rather than the historically used analyses based on available redundant and diverse safety systems with assumed single failures. Furthermore, in the DG-1199 Regulatory Analysis this <u>paradigm shift</u> is not even discussed, implying that this is little more than one of a number of clarifications. Finally, the analyses justifying and applying this treatment to BWR MSIV leakage, and generating recommended parameters, is not complete in the evaluation of pertinent system responses in a manner that would allow a complete need assessment and Regulatory

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				Analysis. The need for consideration of primary containment bypass leakage for isolation valves connected to reactor vessels, for design basis accident assessments, would be inherently a generic issue. It should be considered in a fully risk informed manner. RECOMMENDATION This regulatory guide revision should not be issued with the proposed MSIV guidance, given (1) the issues of incompleteness of the justification for the proposed MSIV dose assessment approach; (2) the unexamined generic implications; and (3) the absence of an appropriately complete regulatory analysis.
95	A-6	Section A-5.1 (3rd para)		COMMENT and RECOMMENDATION: Material from Reference A-10, if used, should be incorporated into the Regulatory Guide.
96	A-6 A-9	Section A-5.1 Reference A-10		COMMENT: The direction and development of Reference A-10 does not take into account the historical basis and reasoning in the development of existing guidance on MSIV leakage dose assessment, which can be summarized as follows: 1. Some early BWRs treated MSIV Leakage as part of LA. MSIV leakage specific leakage limit of 11.5 scfh per line, 46 scfh total. 2. Some early BWRs excluded MSIV leakage from LA, with NRC eventually requiring an application for a 10CFR50 Appendix J exception, and MSIV

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				specific LOCA dose contribution assessment. 3. In 1976, NRC published RG 1.96 R1, requiring MSIV Leakage Control Systems for plants with Construction Permits after March 1, 1970. These complex systems generally direct MSIV leakage to SGTS or back to containment. 4. In 1983, NRC identified MSIV leakage as a generic issue, because of the number of failures to meet above limit. 5. In June 1986 NUREG/CR-4330 identified MSIV LCS as having only a marginal risk-benefit, recommending elimination of requirement. 6. Also in 1986, NUREG-1169 evaluated benefits of alternative MSIV leakage mitigation, chiefly involving balance of plant systems. 7. In 1993, NRC approved GE NEDC-31858P as an acceptable evaluation and basis for use of the turbine-condenser system to mitigate MSIV leakage effects, and further, to allow higher MSIV leakage limits. Included seismic ruggedness evaluation requirements. 8. RG 1.183, R0 endorsed NEDO-31858P as providing acceptable methodology for use in Alternative Source Term based LOCA evaluations. **The above processes represent an initial deterministic approach for MSIV leakage,**

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				followed by an orderly, risk-informed generic issue resolution. RECOMMENDATION: Any changes in MSIV leakage accident analysis treatment should result from a similar (though shorter in time) interactive process that represents lessons learner, evaluation of risk-benefits of the proposed changes. A rigorous, risk-informed evaluation should be applied here, not short-circuited by narrow "backfit" rationalization.
97	A-6	Appendix A Section A-5.1	The third paragraph of this section states that licensees must go to Reference A-10 for specific information from tables and sections required to perform the analysis.	COMMENT: There are several tables from Reference A-10 that are needed in order to comply with this appendix. Although some Table values are quoted in footnote #3, Section 5.2 of the reference is still needed. The source documents for this reference need to be provided in order to perform an adequate review of the DG. RECOMMENDATION: These tables should be provided in this guide to minimize confusion and provide all of the important information needed for the analysis. Provide the source documents for industry review. Additional review time after these documents become available will be required.
98	A-6	Appendix A Section A-5.1	The fourth paragraph of this section states; ***"For BWR designs other than those discussed above, other models of MSIV source concentration will be considered on a case-by-case basis."***	COMMENT: There is specific information provided for BWR Mark I, II, and III. However, there is no such specific information provided for any new reactor designs. RECOMMENDATION:

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				If this guide is also applicable to new reactors, there should be some indication provided with regard to these other models, not just on a case-by-case basis.
99	A-6	Appendix A Section A-5.4	This section states; ***"Reduction in drywell radioactivity due to operable containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited on a case-by-case basis."***	COMMENT: See comments for Section A-5.1 above. ECCS injection system operation should be permitted as designed (not to prevent the fuel damage, but rather to mitigate the EFFECTS of the damage) if operable per Tech Specs and qualified per SRP 6.5.2. RECOMMENDATION: Perform the MELCOR runs using ESF systems in the way that they were designed. A specific run to develop the source term is not needed since one already exists (NUREG-1465). A new run using more accurate plant data (e.g., being able to run ECCS systems when they are designed (and tested) to function). The Sandia MELCOR report appears flawed and should be re-evaluated to allow credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public comment period for DG-1199. Additional technical comments will be available

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				for a future Regulatory Guide revision.
100	A-7	Appendix A Section A-5.6	This section states; "***The piping segments and physical barriers are to be designed, constructed, and maintained to Quality Group A and Seismic Category 1 of ASME Section III requirements (A-11) or have been evaluated to be rugged as described in Regulatory Position A-5.7. The amount of reduction will be evaluated on an individual case basis***"	COMMENT: Appendix C (CRDA) assumes a reduction of iodine concentration in the condenser without regard to its seismic ruggedness for a period of 24 hours. This is also to imply that the steam lines (also without regard to the seismic qualifications) are also credited for the "delivery" of the steam to the condenser (no deposition is credited in the steam lines). The recent earthquakes in Japan (well beyond those considered to be SSE) have shown that no failures of secondary systems/structures resulted. Therefore, it stands to reason that licensees should be able to assume at least some condenser credit for at least the first 24 hours (consistent with the CRDA). Also, now with such a great need for additional credit, approval on an individual case basis does not seem efficient for NRC review or increased RAI questions. RECOMMENDATION: If seismic ruggedness evaluations are required and performed, then all such components and structures should be able to be credited for the entire duration. This could include all steam piping, all appropriate condenser surfaces, all appropriate portions of the main turbine. Since most BWR submittals include some degree of this credit, provide additional guidance to minimize the need for evaluation on individual case bases.
101	A-7	Appendix A Section A-5.8	This section refers to Table 6-2, Table 6-1, and Section 6.3 of Reference A-10 (the Sandia Report).	COMMENT: These important tables should be part of the RG without having to go to another document.

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				RECOMMENDATION: Provide these tables in the DG, not just a reference to an external document.
102	A-7	Appendix A Footnote #4	This section states; ***"A removal coefficient of 0.0 hr^{-1} should be used for the removal coefficient for the in-board piping as described in the footnotes for Tables 6-1 and 6-2 of Ref. A-10."***	COMMENT: Many previous submittals have included this piping (in at least 3 out of four lines) in their deposition models previously approved by the NRC staff. What is the technical justification for this prohibition if turbulence can be shown not to be a problem for deposition? RECOMMENDATION: Provide a technical justification.
103	A-9	Reference A-10		COMMENT: Based on the data presented in Reference 10 for steam dome temperature, it does not appear that the temperature of the steam dome would be sufficiently low relative to that of the steam lines to produce a steam-hydrogen-fission gas mixture with a density greater than that of the steam in the lines. Accordingly, mixing between the steam dome and the steam lines may not be very efficient. The absence of efficient mixing (in concert with activity deposition along the leak path) may produce a large fission product concentration difference between the steam dome and the portion of the steam lines adjacent to the inboard MSIVs. Given this condition, considering the drywell as the source (even with credit for drywell sprays but without credit for steam line deposition up to the inboard MSIVs) may produce a more conservative dose result than using the actual steam dome/steam line pathway.

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				RECOMMENDATION: If this is the case, then the requirement for activity concentration adjustment factors and the recommendation against credit for drywell sprays should be deleted from the final regulatory guide.
104	B-2	Appendix B Section B-2 See also Comment 111	This section states; "***...(i.e., 99.5 percent of the total iodine released from the damaged rods is retained by the water) may be assumed. The difference in DFs for elemental (99.85%) and organic (0.15 percent) iodine species results in the iodine above the water that is composed of 70 percent elemental and 30 percent organic species.***"	COMMENT: Since no reference given, what is the technical justification for this change? RECOMMENDATION: Provide technical justification.
105		Appendix B Section B-2		COMMENT: The NRC endorses setting the effective spent fuel pool factor (DF) to 200 under certain conditions. The envelope over which this endorsement holds has been revised by adding the condition that the release pressure not exceed 1,200 psig. Currently, the NRC staff endorses setting the spent fuel pool DF for diatomic iodine to 500. Given this and the assumed iodine composition fractions (unchanged from RG 1.183), this condition seems unnecessarily restrictive. RECOMMENDATION: The envelope over which the spent fuel pool effective DF may be set to 200 should include release pressures up to 1300 psig. Studies using Westinghouse methodologies (WCAP-7828) can be applied to support a pool DF of

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				500 for diatomic iodine for fuel pin release pressures up to 1300 psig. Given the iodine chemical composition fractions endorsed here, this provides additional margin for applying an effective DF for fuel pin release pressures up to 1300 psig.
106	B-3	Appendix B Footnote #4	This footnote states; ***"Technical specifications that allow such operations usually include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes."***	COMMENT: The "requirement" to close the containment openings within 30 minutes is still included. Being as how most licensees are being granted up to 1 hour to close the containment, why not change the RG to 1 hour to relax this? RECOMMENDATION: Change to 60 minutes instead of 30 minutes.
107	B-2	Appendix B Section B-4.1	This section states; ***"The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period. The release rate is generally assumed to be a linear or exponential function over this time period."***	COMMENT: This causes release rates to the environment that are extremely high. Plants typically do not have air handling equipment capable of performing this magnitude of release. It also makes one to believe that the safest place to be at t=2 hours after a FHA is on the fuel floor since all activity has been released to the environment (or the control room). This, coupled with the 50% mixing statement in Section B-5.3 is grossly overly conservative, causing concerns with many licensees. RECOMMENDATION: Relax the release requirements to something more realistic. This may be a topic for discussion in a future NRC/Industry Task Force Meeting to resolve.
108	C-1	Appendix C	A BWR CRDA is identified in Regulatory Position 3.2 as a reactivity initiated accident.	Comment: This Appendix does not refer to the accident as a RIA. The use of the delta-h for RIAs is

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				also not mentioned. RECCOMENDATION: This section should also refer to it as a RIA for consistency. It should also discuss how the delta-h is used.
109	D-1	Appendix D Section D-3	The fact that "***Noble gases should be assumed to enter the steam phase instantly***" is said twice in this paragraph.	COMMENT (Editorial): See the second and fourth duplicate sentences. RECOMMENDATION: Recommend removing one of the duplicate sentences.
110	E-1	Appendix E Section E-1	Regulatory Position 3.6 of this DG states; "***For the postulated main steamline break, steam generator tube rupture, and locked rotor accidents, the licensee should evaluate the amount of fuel damage assuming that the highest worth control rod is stuck at its fully withdrawn position.***"	COMMENT: This section does not state that the highest worth control rod is stuck in it's full out position. However, Section 3.6 now requires this for all non-LOCA accidents (except FHA & BWR CRDA). RECOMMENDATION: Resolve the inconsistency between the two sections.
111	D-2	Appendix D Section D-4.4	This section states; ***"The iodine species released from the main steamline should be assumed to be 95 percent cesium iodide as an aerosol, 4.85 percent elemental iodine, and 0.15 percent organic iodide."*** However, RIS 2006-04 states; ***"For some accidents (e.g., main steamline break and rod drop), licensees have excluded noble gas and cesium isotopes from the dose assessment. The inclusion of these isotopes should be addressed in the dose***	COMMENT: Source terms for MSLB are defined on page D-1. The discussion of cesium iodide in section 4.4 confuses the issue. This item was discussed as an issue in RIS 2006-04, but no additional guidance appears to be provided here. The NRC presentation at the November 16, 2009 Workshop specifically stated that the proposed DG included items from RIS 2001-19 and RIS 2006-04. However, neither of these is listed as a reference. RECOMMENDATION:

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			assessments for AST implementation."	Provide additional guidance. As a minimum, delete the reference to Cesium Iodide and refer to it only as an aerosol. These types of statements in the RG are intended to address the chemical form of iodines, not source terms.
112		Appendix E & all pages following Page E-1	In Appendix E § E6.4 and Appendix G § G5.4 (see notes in the left hand column on the mislabeling of these sections), the NRC staff states that "The release of fission products from the secondary system should be evaluated with the assumption of coincident loss of offsite power."	COMMENT: This guideline could be interpreted to absolve licensees from having to analyze accident scenarios with offsite power maintained even though a higher release of activity could be computed for this scenario. This interpretation conflicts with the guideline in the 5th sentence of § 5.1.2 in which the staff states that "Assumptions regarding the occurrence or timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences." Finally, it is noted that this guideline does not appear in appendices associated with other accidents presumably followed by releases from the steam generators (SGs). RECOMMENDATION: It is assumed at present that the NRC staff intends licensees to make assumptions pertaining to offsite power to "maximize the postulated radiological consequences." In this case, it is recommended that the staff strike the subject passage from the guidelines in Appendices E and G. If the staff desires to retain this passage, it is recommended that they amend it to state "Assumptions regarding the occurrence or timing of a loss of offsite power should be selected with the objective of maximizing the calculated release of activity from the SGs" or equivalent. If the staff

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				chooses this latter recommendation, they may want to consider inserting this guideline as amended in Appendices F and H as well as Appendices E and G. On the other hand, the staff may really intend to absolve licensees from having to consider accident scenarios with offsite power available in evaluating "the releases of fission products from the secondary system." If this is their intent, the staff should positively and definitively state this and ensure that it is inserted in Appendices E-H.
113		Appendix E, § E-6.6.1, 2nd bullet, 2nd sentence § E-6.6.2; Appendix F, § F-6.6.2, lead paragraph, 2nd sentence, & § F-6.7		COMMENT: The 2nd sentence of the 2nd bullet of § E-6.6.1 provides guidance for the scenario of SG tube bundle uncovery. The only guidelines in § E-6.6.2 pertain to post accident SG tube bundle uncovery. Having guidelines for post accident SG tube bundle uncovery in separate sections may lead to confusion. This remark also pertains to the 2nd sentence of the lead paragraph of § F-6.6.2 and § F-6.7. RECOMMENDATION: The staff may want to consider moving the 2nd sentence of § E-6.6.1 to § E-6.6.2. Likewise, the staff may want to consider moving the 2nd sentence of the lead paragraph of § F-6.6.2 to § F-6.7.
114		Appendix E, § E-6.6.1, 2nd bullet, 2nd sentence § E-6.6.2;		COMMENT: The staff guidelines cited in these passages apparently have licensees calculate only a flash fraction for SG tube leakage during intervals of postulated SG tube bundle

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		Appendix F, § F-6.6.2, lead paragraph, 2nd sentence, & § F-6.7		uncovery. In particular, the guidelines apparently do not encompass the potential for atomization of the SG tube leakage and bypass of at least some of the atomized droplets. In the past, the staff has identified the process of atomization of SG tube leakage and subsequent bypass during time spans of SG tube bundle uncovery as separate from flashing (NUERG-0800 Rev 2 & § 15.6.3.III.10). <u>RECOMMENDATION:</u> The staff may wish to replace "flash" with "combined flash and atomization bypass" or equivalent in the 2nd sentence of § E-6.6.1. In like manner, in the 2nd sentence of the lead paragraph of § F-6.6.2, the staff may want to replace the clause "a portion of the primary-to-secondary leakage will flash to vapor" with "a portion of the primary-to-secondary leakage will either flash to vapor or be atomized with some of the atomized droplets exiting the SG(s) with steam flow" or equivalent.
115	38	References	RIS 2001-19 and RIS 2006-19 were quoted in the 11/16/2009 NRC Workshop. However, the reference section does not include references to these	<u>COMMENT:</u> Although these were stated in the Workshop, they are not referenced in the DG. It is also not clear that all issues have been revised within the DG. <u>RECOMMENDATION:</u> Include RIS 2001-19 and RIS 2006-19 as references. Ensure all issues have been addressed satisfactorily.
116	28	C Section 5.1.2	Section 5.1.2 states; ***"Assumptions regarding the occurrence and timing of a loss of offsite power should be***	<u>COMMENT:</u> Although Section 5.1.2 appears generic, possibly pertaining to all accidents, only

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		Section E-6.4 Section G-5.4	***selected with the objective of maximizing the postulated radiological consequences."*** Sections E-6.4 and G-5.4 state; ***"The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power."***	Appendices "E" and "G" make similar statements. This adds confusion since none of the other accident appendices include this statement. RECOMMENDATION: Either include the statement in all accident appendices or remove it from Sections E-6.4 (Page E-2) and G-5.4 (Page G-2).
The following comments pertain to PNNL-18212				
P-1		General Comment		COMMENT: The work in PNNL-18212 is based in part on an application of a proposed revision to the ANS 5.4 standard. As noted in this report (cf. Ref. 2 Footnote on Page 2), the proposed revision to ANS 5.4 is still under review within the ANS with approval not expected before 2010. To date, this proposed revision is not available within the industry. RECOMMENDATION: Since it is a basis for the analysis reported in PNNL-18212, we recommend that the staff consider delaying making revisions to DG-1199 based on PNNL-18212 until the ANS approves the draft revision to ANS 5.4 and the authors of PNNL review the approved revision to verify that the work reported in it remains valid given any changes between the draft revision to ANS 5.4 used as a basis for that work and the approved version. Furthermore, the draft revision to ANS-5.4 should be reviewed within the nuclear industry.
P-2		Page 2, 2^{nd}	This sentence states that "The reason for updating	COMMENT:

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		paragraph, 3rd sentence.	the ANS-5.4 and the Table 3 values in Regulatory Guide 1.183 is that the 1982 ANS 5.4 standard significantly over predicts the iodine release fractions."	The non LOCA gap fractions for the iodine isotopes except ^{132}I remain unchanged. It is proposed to increase the non LOCA gap fraction for ^{132}I from 5% to 23%. <u>RECOMMENDATION:</u> The results of PNNL-18212 appear to support small decreases in the gap fractions for halogens other than ^{132}I. In particular, we propose non LOCA gap fractions of 7.5% for ^{131}I and 4.5% for all other halogens other than ^{132}I. As an alternative, the authors should strike the above mentioned sentence and state instead that the non LOCA gap fractions are revised to accommodate nodal power and burnups outside the envelope defined in RG 1.183 Footnote 11.
P-2		2nd paragraph		<u>COMMENT</u>: FRAPCON does not account for daughtering after shutdown. By 2 hours only 55% of the original ^{132}I exists, by 8 hours only 9%. The additional source (birth) of ^{132}I is from the daughtering of ^{132}Te and should be accounted for else the ^{132}I release to birth ratio (i.e. from the lack of daughtering) will be larger than it should be. Because release to birth is an artifact of the method and does not account for daughter effects the FRAPCON model should only be reasonable for those isotopes whose birth does not significantly come from the decay of a parent isotope, for example I-131, I-133, I-134, and I-135. <u>RECOMMENDATION</u>: The increased value for the release of I-132 should be changed back to the value for the other halogens

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P-4		Page 2, 3rd paragraph, last sentence.		COMMENT: Here and in other passages in PNNL-18212 the values for the non LOCA gap fractions (both the baseline values of Table 3 and the post REA values of Table 4) correspond to a "95/95 tolerance level." This is a significant departure from the current NRC guidelines for the treatment of variances and uncertainties developing radioactive source terms for design basis accidents. For example, the release fractions specified for the LOCA in RG 1.183 Tables 1 and 2 evidently are mean values. (The corresponding "Early in-vessel" release fractions of Tables 3.9 and 3.10 of NUREG-1465, the basis document for RG 1.183 are described therein as "Mean Values.") Taking mean values for release fractions in defining radioactive source terms does not make the analyses of design basis accidents "non conservative." Rather, these analyses already incorporate several layers of conservatisms. Limiting values are taken for all parameters pertaining to transport of fission products, their release to the environment, and their accumulation in the control room. For example, if a parameter is governed by a technical specification (containment leak rates, ventilation filter penetration test criteria, etc.), the limiting value in that technical specification is taken, sometimes after applying additional margin (for example, a factor of 2 is applied to the ventilation filter test criteria). The values calculated for atmospheric dispersion factors for the exclusion are boundary are 99.5th percentile values (cf. RG 1.145), while those

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				for the boundary of the low population zone and control room are 95th percentile values (RG 1.194). These are examples of conservative margin already inherent in existing AST analyses. In light of them, it seems unnecessary to depart from the current practice in RG 1.183 of taking mean values for release fractions. <u>RECOMMENDATION:</u> Consider taking mean values for non LOCA gap fractions in issuing Tables 3 and 4.
P-5		Page 4, 3rd paragraph; Page 5, 2nd paragraph, last sentence; Page 9, 4th complete paragraph.		<u>COMMENT:</u> This remark is made to provide an example for the evident need for a review of the ANS-5.4 standard. It is noted in the passages cited above that the diffusion coefficient for ^{85}Kr, a noble gas, is set to a lower value than the diffusion coefficients for the isotopes of other chemical elements. Given their completed valence electron shells, noble gases normally do not react with other chemicals. In particular, ^{85}Kr will not bond chemically with atoms or molecules in the nuclear fuel matrix as the more reactive isotopes may. It stands to reason that its diffusion coefficient would take a higher value than those of isotopes of more reactive elements. As an example in the extreme, cesium is one of the most reactive elements in nature and normally does not occur in the form of an element. Yet it diffusion coefficient is set to twice the value of that of ^{85}Kr. <u>RECOMMENDATION:</u> The staff should promote the validation of the

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				methodology of ANS-5.4 given the chemical properties of the fission products presumably in the fuel pin gaps. In particular, the assumptions concerning the diffusion of these fission products should be validated. As needed, the methodology of ANS-5.4 should be revised to take into account these chemical properties.
P-6		Page 6, § 2.2.4, 2nd paragraph, 7th sentence.		COMMENT: This sentence alludes to increasing burnup at constant power. In general, fuel pin power decreases with increases in burnup. RECOMMENDATION: PNNL could consider reviewing this sentence. The statement may be a reasonably conservative assumption but if so, should be so stated.
P-7		Page 8, § 3.1, 2nd paragraph.		COMMENT: This paragraph and the formula following it give the increase in gap fraction for a rod ejection accident (REA) as a linear function of fuel enthalpy increase. It is not clear that this function would be linear. RECOMMENDATION: Consider stating the basis for assuming a linear relationship between post REA fuel enthalpy rise and increase in post REA gap fraction.
P-8		§ 3.1, paragraph spanning Pages 8 & 9, cf. Figure 6 (post REA		COMMENT: This passage evidently implies that the three data points (HBO-2, HBO-3, and HBO-4) above the lines in Figure 6 are anomalous. RECOMMENDATION:

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		fission product release fractions versus peak enthalpy increase).		Consider removing these points from the data sets and recalculate the "upper 95/95" fits.
P-9		Figure 6 (cf. Page 9, 2nd complete paragraph).		COMMENT: A data point (associated with the CABRI tests) is associated with mixed oxide (MOX) fuel. The NRC states that the post accident release fractions (RG 1.183 Tables 1-3) "may not be applicable to cores containing mixed oxide (MOX) fuel [cf. Footnote 10]." The NRC has retained this guideline in DG-1199 (cf. Footnote 11). RECOMMENDATION: All data that are associated with MOX fuel should be removed from the data set pertaining to post REA release fractions versus peak fuel enthalpy increase and the "upper 95/95" fits recalculated. As an alternative, the NRC staff could issue guidelines for all post accident release fractions that apply to MOX fuel (counterparts to Tables 1-4).

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P-10	19	Tables 3 and 4		COMMENT: Tables 3 and 4 Pages 19 & 20 of DG 1199 indicate that the release of alkali metals is a minimum of 46%. This appears to be excessive. There appears to be no confirmatory evidence for the use of FRAPCON when applied to non noble gas release. The FRAPCON 3 model has validation for the release of noble gases and the FRAPCON 3 model may be valid for release of iodine and cesium from within the grain to the grain surface. Transport from the grain surface to gap is treated in FRAPCON 3 as if iodine and cesium were noble gases, i.e. saturation of the pore equals release to the gap and no holdup during a RIA. However, both iodine and cesium are highly reactive and should readily combine with each other to form condensed phase CsI. In addition cesium should combine with uranium, and molybdenum to form other condensed phases that are also not easily transported. The chemistry models used within the NRC VICTORIA code indicate that this readily happens. FRAPCON does not account for chemistry effects. RECOMMENDATION: NRC should confirm that FRAPCONhas been compared with the ORNL VI experiments, the SNLA ST experiments, or the Phebus/VERCORS RT-6 tests to see the effect of different temperatures and atmospheres on the release of non-noble elements like iodine and cesium? FRAPCON

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				should also be compared with the total rod release Booth diffusion models within the NRC MELCOR code. A temperature transient as determined by FRAPCON should be used as input to the NRC VICTORIA code to examine chemical effects.
The following comments pertain to MELCOR SAND 2008-6601				
M-1	A-9	Reference A-10		COMMENT: The test procedure used by the facilities substantially over predicts the leak rate thru the MSIV's. Hence, shouldn't an acceptable model start with a best estimate model and then do appropriate variations? Reference A-10 uses the term "best estimate" 15 times. The term best estimate applies to an approach where code input, models, options, and conditions are prepared or developed with no known bias toward conservatism (T.S. Kress),

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				yet the approach taken in the referenced analysis clearly does not meet this definition. RECOMMENDATION: In the final guide, NRC should better clarify why and how the approach (from Reference A-10) has been incorporated as the technical basis for the proposed changes to the MSIV leakage pathway in regard to the concepts, "best estimate" and "reasonably conservative," that have shaped the development and use of alternate source terms up to now.
M-2	A-6	A-5.1	General comment	RECOMMENDATION: The need for activity concentration adjustment factors to account for higher activity concentration in the steam dome (as compared to the drywell) when calculating the dose contribution from MSIV leakage should be reevaluated. The reevaluation should take into account the potential for stratification and suppressed mixing between the steam dome and the steam lines from the reactor vessel to the inboard MSIVs. Credit for drywell sprays should be allowed, as well.
M-3	A-6	Appendix A Section A-5.1	The first paragraph of this section states; "The source of the MSIV leakage is assumed to be the activity concentration in the reactor vessel steam dome. At the end of the early in-vessel release phase, the activity concentration in the vessel dome should be assumed to equal the containment (or drywell) activity concentration."	COMMENT: Fuel damage was determined in the Sandia MELCOR report assuming a 2-hour lack of ECCS injection (i.e., re-flood). However, considering all of the system redundancy and quality designed into the plant, these systems WILL function as designed. If we are still assuming the required significant core damage from Tables 1 and 2 as required, why can't we assume that appropriately qualified ECCS components will still function to mitigate the effects of this significant core damage (the

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				amount of core damage remains unaffected in spite of any mitigative actions)? Section A-5.4 allows credit for operable qualified systems on a case-by-case basis. If this is the case, why can't appropriately qualified ECCS injection systems be credited to mitigate the effects (in their designed time frame), not altering the source term being released from the fuel? Also see comments and questions for Section 5.1.2 in the main body of the document. RECOMMENDATION: Perform the MELCOR runs using ESF systems in the way that they were designed. A specific run to develop the source term is not needed since one already exists (NUREG-1465). A new run using more accurate plant data (e.g., being able to run ECCS systems when they are designed (and tested) to function). The Sandia MELCOR report appears flawed and should be re-evaluated to allow credit for designed and qualified ESF functions using an assumed non-mechanistic severe accident source term. Therefore, it is recommended that this analysis be re-performed. A detailed review of the MELCOR report using design basis parameters, response times, and functions may result in recommended changes, corrections and conclusions in the Sandia MELCOR report. If performed by industry (potentially by the BWROG), this work will most likely not be funded, approved, or concluded before the end of the public comment period for DG-1199. Additional technical comments will be available for a future Regulatory Guide revision.

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M-4	A-6 A-9	Section A-5.1 Reference 10	General treatment of Turbine-Condenser system response	COMMENT: The original NEDC-31858P MSIV treatment represents a relatively simplified and conservative treatment of MSIV leakage. The MELCOR analyses is confined by treating the turbine-condenser response as being simplified in NEDC-31858P. These simplifications include: 1. The steam line pathway was assumed to result in releases largely through HP turbine gland seals with gravitational settling and deposition in steam line piping that has been evaluated for seismic ruggedness. No credit is taken at that time for HP turbine holdup, or transport to downstream piping and systems, potentially including the condenser. 2. The condenser pathway, if used, was credited once a sufficient path is established, with this pathway, the condenser, and the surrounding turbine building credited if determined to be seismically rugged. Releases from this path were assumed to be largely through LP turbine seals. Credit is taken for holdup, gravitational settling and deposition, with release to the condenser below the condenser cooling tubing. In general, no credit is taken for deposition on the available massive surface area of these tubes, nor for questioning the applicability of aerosol releases that must be transported up to the likely release

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				point. The concept of consideration of seismic ruggedness of BWR turbine-condenser resulted from a detailed review of the response of these types of systems to actual earthquakes. This data is discussed in NEDC-31858P, and is continuously maintained and updated in the EPRI earthquake experience database. This historical data and recent experience suggest that BWR turbine condenser systems would remain largely intact, and that they could be modeled as such. MSIV leakage through pathway 1 to a largely intact turbine condenser system would be transported slowly through main steam lines to the HP turbine volume. Flow from the HP turbine could, in part, be released through turbine seals, but may largely continue to downstream systems, depending of pressure conditions. RECOMMENDATION: 1. The MELCOR model used in the A-10 evaluation should be updated to reflect the entire turbine condenser system, through the LP turbine and condenser for scenarios where the combined intermediate valves are open. The MELCOR model used in the A-10 evaluation should be updated to reflect in impact of condenser tubing, above and below zones, and possibly the impact of shell as well as LP Turbine Shell Leakage.

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M-5	D-2	Appendix D Section D-4.1	This section states; "The main steamline isolation valves should be assumed to close in the maximum time allowed by technical specifications."	COMMENT: Although this has nothing to do with this section/accident, the Sandia MELCOR report indicated that the inboard MSIV closes in 3 seconds and the outboard MSIV closes 3 seconds later. This puts the outboard valve closure at 6 seconds, which is LONGER than the Tech Spec allowed value. RECOMMENDATION: The Sandia MELCOR report appears to be incorrect and needs to be corrected. Perform the MELCOR runs using these systems in the way that they were designed and approved.
M-6	A-6	Appendix A Section A-5.8	"Section 6.3 of Reference A-10 describes an acceptable model for estimating the aerosol deposition in horizontal piping. From the start of the accident to the termination of the early in-vessel release phase, the amount of reduction in the steam line is determined by the removal coefficients in Table 6-2 of Reference A-10. After the early in-vessel release phase ends, the removal coefficients are given by the values in Table 6-1 of Reference A-10."	COMMENT: MELCOR (reference A_10) does not contain a decay and daughtering algorithm as it does not attempt to determine nuclide variation in time. Nuclides like Kr-87 that decay into Rb-87 in 76 minutes and release a substantial amount of energy are allowed to remain in the decay heating equation for the entire transient as the model is based only on the initial inventories. Thus releases that extend for 10 hours would continue to decay Kr-87 when in reality there would only be less than 1% of it available. Similarly Kr-88 also releases a large gamma with a 170 minute half life so less than 10% exists at 10 hours. MELCOR does not include a chemistry model hence it revolatilizes elements instead of species. For example Cs goes volatile at 1000 K but CsI is at 1500 K and the hydroxide (CsOH) at 1250 K and the molybdate at 2500 K.

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				RECOMMENDATION: Use other NRC codes to evaluate revolatilization of radionuclides in the system, perhaps including a few more chemical classes in MELCOR would help resolve part of the issue. Also add a least a simple decay term to the decay heating model so that it is based on current levels of radionuclides and not those at the start of the accident.
M-7	A-6	Appendix A Section A-5.9	"Reduction of the amount of released elemental iodine by plateout deposition on steam system piping may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. The model should be based on the assumption of well-mixed volumes. Reference A-13 provides guidance on an acceptable model."	COMMENT: MELCOR (reference A-10) contains it's own deposition/revolatilization modeling. RECOMMENDATION: Use Ref A-13 to validate the MELCOR models, or A-10 to validate A-13.

Note 1

1) This note provides details for an alternative resolution of (Comment 76 – Will update comment reference in final version]. This note proposes a generalized set of guidelines for assigning the 0-2 hour control room χ/Q value to the 2 hour time span of the limiting release of radioactivity to the environment. The alternative resolution consists of replacing the last two sentences of the 2nd paragraph of § 5.3 with the following or equivalent:

"Recommendations are made for two cases. In the first case, the limiting 2 hour interval falls completely within an averaging period $T1$-$T2$ for the control room χ/Q as shown in Figure 1 below

Figure 1

$T1$ ——— $T2$

0-2hr

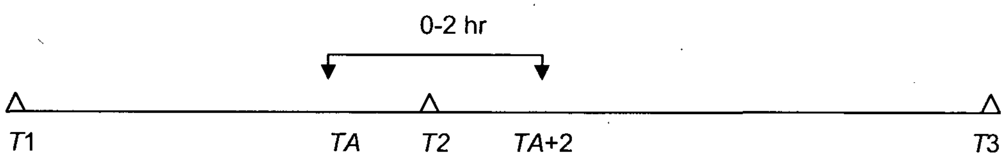
The 0-2 hour control room χ/Q value $(\chi/Q)_{(0\text{-}2)}$ is assigned to the limiting 2 hour interval and the "($T1$+2)-$T2$" value $(\chi/Q)_{[(T1+2)\text{-}T2)]}$ assigned to the remaining $T2 - T1 - 2$ hours of the interval $T1$-$T2$. If not calculated with ARCON96, the "($T1$+2)-$T2$" control room χ/Q value may be calculated using the following relation (1):

$$\left(\chi/Q\right)_{[(T1+2)-T2]} = \frac{(T2 - T1)\left(\chi/Q\right)_{[T1-T2]} - 2\left(\chi/Q\right)_{0-2}}{T2 - T1 - 2},$$

Here $(\chi/Q)_{[(T1+2)\text{-}T2)]}$ is control room χ/Q value over the averaging interval $T1$-$T2$. If (1) yields $(\chi/Q)_{[(T1+2)\text{-}T2)]} = 0$, then $(\chi/Q)_{[(T1+2)\text{-}T2)]}$ should be set to 0.

In the second case, the limiting 2 hour interval ["TA-(TA+2)"] overlaps two averaging periods $T1$-$T2$ and $T2$-$T3$ as shown in Figure 2 below:

Figure 2



In Figure 2, the 2 hour of maximum releases begins at TA and continues between the end point T2 between the averaging intervals $T1$-$T2$ and $T2$-$T3$ so that $T1 < TA < T2 < TA + 2 < T3$. The 0-2 hour control room χ/Q value is assigned to the intervals TA-$T2$ and (TA+2)-$T3$. Given that the control room χ/Q values for the remaining partitions are not calculated with ARCON96, they may be obtained as follows. The control room χ/Q value for the interval $T1$-TA is obtained using (2)

$$\left(\chi/Q\right)_{(T1-TA)} = \frac{(T2 - T1)\left(\chi/Q\right)_{[T1-T2]} - (T2 - TA)\left(\chi/Q\right)_{0-2}}{TA - T1},$$

The control room χ/Q value for the interval (TA+2)-$T3$ is obtained using (3)

$$\left(\chi/Q\right)_{[(TA+2)-T3]} = \frac{(T3 - T2)\left(\chi/Q\right)_{[T2-T3]} - (TA + 2 - T2)\left(\chi/Q\right)_{0-2}}{T3 - TA - 2},$$

If either (2) or (3) returns a negative control room χ/Q value, that control room χ/Q value should be set to 0."

REFERENCES CITED IN THE COMMENTS

1) Memorandum, Anthony Taylor (USNRC Nuclear Performance and Code Review Branch) to Robert Taylor (Chief, USNRC Dose Assessment Branch), "Technical Basis for Revised Regulatory Guide 1.183 Fission Product Fuel-to-Cladding Gap Inventory," February 10, 2009.

2) C.E. Beyer (Pacific Northwest National Laboratory) and P.E. Clifford (USNRC), Update of Gap Release Fractions for non-LOCA Events Utilizing the Revised ANS 5.4 Standard, PNNL-18212, February, 2009.

3) L.Soffer, S.B. Burson, C.M. Ferrell, R.Y. Lee, J.N. Ridgely, Accident Source Terms for Light-Water Nuclear Power Plants, NUREG-1465, June 1992.

4) W.D. Travers, "Proposed Generic Letter 98-XX 'Steam Generator Tube Integrity,'" SECY-98-248, October 28, 1998. The document accession number is ML992920090.

5) USNRC, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP), NUREG-0800. In general, the NRC review expectations in SRP Rev 2 are in the current license bases for Catawba and McGuire Nuclear Stations. The NRC has approved full scope implementation of the method of AST for both Catawba and McGuire. As such, the guidelines of RG 1.183 supplant all NRC review expectations in SRP 15 pertaining to the analyses of radiological consequences of DBAs. None of these review expectations remain with the current license basis for either nuclear plant.

6) USNRC Spent Fuel Project Office, Interim Staff Guidance-5 (ISG-5) Rev 1, Confinement Evaluation.

7) Title 10 of the Code of Federal Regulations, November 17, 2009. This document is denoted henceforth as 10 CFR. The specific section cited here is 10 CFR 72.106(b).

8) USNRC, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors, Regulatory Guide 1.195 (Rev 0), May 2003.

9) USNRC, Assumptions Used for Evaluating Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Regulatory Guide 1.25 (Rev 0), March 1972.

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1	1. The alternative source term methodology described in the draft regulatory guide permits the assumption that the release of radioactive effluent to the environment occurs at some time period following the onset of the accident within the plant facility. Section 5.3, Meteorology Assumptions, provides guidance on pairing atmospheric dispersion factors (c/Q values) with the periods of maximum to the environment. a. Is it equally or more appropriate to include consideration of engineering factors such as time of control room isolation and initiation of filtration, in addition to the time sequence release of radiological effluent to the environment, when assessing the limiting dose to control room operators?	(See also the specific comments related to Section 5.3 of DG-1199 provided in Attachment 1). The industry understands the focus of the question to refer to the application of the 0-2hr X/Q to the 2 hr period during which the maximum hypothetical dose to personnel in the control room can be calculated. It is not expected that including consideration of engineering factors, such as is outlined in the question, would produce any appreciable difference in an assessment of postulated operator doses in comparison to applicable general design criteria because control rooms are typically put into protective modes either by automatic actuation or manual operation prior to or within the period of maximum release. It is felt that inclusion of consideration of engineering factors and the burden of repetitive evaluations would not be justified by the potential for a marginal refinement in the results, when the overall dose assessment methodology already contains a number of other conservatisms.
2	2. Table 3 of DG–1199 provides revised non-loss of coolant accident fission product gap inventories applicable to all current fuel designs. The purpose of revising Table 3 was to expand its applicability by replacing the prior footnote 11 limitation (*i.e.,* 6.3 kw/ft beyond 54 GWd/MTU) with bounding fuel rod power envelopes. a. Does the bounding fuel rod power envelopes depicted in Figure 1 of DG– 1199 provide sufficient fuel management flexibility such that current and anticipated fuel loading patterns will be able to utilize the Table 3 fission product gap fractions? b. Fission gas release and the resulting fission product gap	(See also the specific comments on Table 3 of DG-1199 provided in Attachment 1) General Comment: The power versus exposure curves presented in the DG-1199 do not envelope reactor operations currently. supported by some vendors. Since the NRC/PNNL claim the release fractions presented are conservative the curves should be removed and only a footnote should be included within the guide indicating that the presented gap release fractions are bounding to peak fuel rod average exposures up to 62 GWd/MTU. The Regulatory Guide should <u>NOT</u> preclude the possibility for fuel rod average

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	inventory are sensitive to fuel rod design and rod power history. To maintain consistency with current regulatory guidance, the revised Table 3 remains applicable to all current pressurized water reactor (PWR) and boiling water reactor (BWR) fuel rod designs (limited only by the bounding power envelope). Significant reductions in fission product gap inventories are achievable with specific fuel rod design calculations (*e.g.,* PWR 17×17 versus PWR 14×14) and/or less bounding rod power histories. Should RG 1.183 provide alternate versions of Table 3, each with its own set of applicability criteria?	exposures beyond 62 GWd/MTU in the future. A path and/or acceptable method for calculating approved source terms at higher burnup should be included in the Regulatory Guide. (i.e., the Regulatory Guide should not constrain the maximum allowable fuel exposure). Comment in Response to Question 2.a: The bounding fuel rod power envelope for BWR fuel in Figure 1 of DG-1199 does not provide sufficient fuel management flexibility for the some current fuel products. In addition, the increases in release fractions contained in DG-1199 could reduce fuel management flexibility by reducing margins for licensees that can meet less bounding power histories. Comment in Response to Question 2.b: At a minimum, separate gap release fractions should be specified for PWR and BWR. Further specifying gap release fractions by fuel mechanical design (e.g., BWR 10x10) and less bounding power histories would provide further enhancement to RG 1.183. DG-1199 should also provide a method for developing gap fractions vs. power history for supporting licensee amendment requests which may be needed for future fuel mechanical designs.
3	3. Reference 18 of DG–1199 documents the expanded fission gas release empirical database and methods used to calculate the revised Table 3 and Table 4 fission product gap inventories. Are any further fission gas measurements available which would help enhance the gap inventories listed in Table 3 and 4?	The model should be compared with at least the PHEBUS/VERCORS RT-6 data and should also yield results reasonably consistent with those from the best estimate MELCOR rod release model. If possible, a temperature transient should be run thru the VICTORIA code to verify the effect of chemistry on release of non-noble gas nuclides.