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Subject: Comments on Draft Regulatory Guide, DG-1113, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors" 67FR3743 dated January 25, 2002.

Duke Energy offers the attached comments relative to the solicitation for public comments regarding Draft Regulatory Guide, DG-1113, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," as published in the Federal Register on January 25, 2002. Duke Energy also fully endorses the industry comments being provided by NEI regarding DG-1113.

Please address any questions to Lee Hentz at 704-382-8081.

Thank you for the opportunity to provide these comments.

Sincerely,

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DUKE POWER COMPANY – DG-1113 COMMENTS & RECOMMENDATIONS

#	Page #	Section, Para #	Comments	Recommend Revisions
1	2, 3, 4, 16	B, C.1.1, C.1.2 C.4.2.4	Various statements in DG-1113 demonstrate the substantial levels of conservatism that are layered into the design and evaluation process to accommodate uncertainties. This level of conservatism is unnecessary and could affect the ability for licensees to reach proper decisions regarding how resources are assigned to maintain and improve the facility. DG-1113 states: • DBA analyses are intentionally conservative in order to compensate for uncertainties in accident progression, fission product transport, and atmospheric dispersion (Section B, page 2) • Defense in depth is an effective means to account for uncertainties in equipment and human performance (Section B, pages 2-3) • System design should incorporate sufficient safety margin to account for analysis uncertainties (Section C.1.1 page 3) and the system design for defense in depth (system redundancy, independence and diversity) must also be conservative to account for uncertainties in accident progression and analysis data (Section C.1.2, page 4). • Delays in actuation due to hold up of radioactivity transport are imposed, but reduction in dose due to transport of activity from fuel to containment release is not credited (C.4.2.4) Realistic evaluations should be used to demonstrate the margin achieved by applying additional conservative assumptions. This new approach should use sensitivity analyses, engineering judgment, and risk-based insights as methods for demonstrating that uncertainties have been considered.	Develop and endorse a new approach that uses realistic evaluations to demonstrate the margin achieved by applying additional conservative assumptions at the conclusion of the analysis rather than on multiple input values or modeling assumptions used in the analysis.

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2	4	C.1.2, 3rd	The last sentence should be clarified because the term "nonconservative results" is not defined and may be misinterpreted. The phrase "radiological and nonradiological" as a modifier to "safety analyses" by its nature encompasses all safety analyses and may be deleted.	Revise the sentence to read: "Radiological analyses generally should be based upon assumptions and input values that are consistent with those used for the correlated design basis safety analysis that provide their input conditions and forcing functions. This should include consistency in physical plant data, operating conditions, as well as event sequences, unless this approach would yield results that would be less severe than realistic evaluation results, or where the approach would become inconsistent with that specified in this guidance document."
3	4	C.1.3.1, 1st	This paragraph imposes a new requirement that each applicable accident listed in this regulatory guide, FSAR, or other licensing documents be evaluated regardless if the accident is part of the current licensing basis. In addition, the text infers that the licensee needs to analyze all listed accidents for each evaluation or plant modification. Traditionally, the NRC has accepted an analysis of the limiting accidents applicable to the proposed change.	Revise the first two sentences to read: "A fundamental commitment required for application of the methodology in this guide is to perform an assessment of each accident applicable to a plant's licensing basis for the proposed change. The licensee shall perform an analysis of those accidents applicable to the proposed change."
4	4, 5	C.1.3.1, 2nd, 3rd	This document does not provide specific guidance regarding acceptable licensing evaluations for any of the examples listed in this section. The examples appear to open the scope of the analyses applications beyond what the remainder of the document covers (or what it should be expected to cover). The scope and intent of this document should be clarified in relation to the examples. The types of analyses that may be performed to address each of these examples may differ widely, and the application of any portion of the guidance that is provided in DG-1113 varies depending on the reasons for and detail of the evaluation.	Modify the second paragraph to read : "There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. A plant's licensing basis may include, but not be limited to, the following." Modify the third paragraph to read: "There may be other areas in which the technical specification bases and various licensee commitments refer to specific evaluations. A plant's licensing basis may include, but are not limited to, the following from Reference 2, NUREG-0737."

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5	5	C.1.3.2	DG-1111 (p.4) states: "Since the existing licensing basis methodology remains valid, a licensee may use the ARCON96 code and the other models addressed in this guide on a selective basis, that is, it is not necessary that all existing χ/Q values be updated at the same time." Similarly, it is appropriate to include the same provision in DG-1113. This will permit licensees to selectively adopt analysis methods features of DG-1113 applicable to the assessment. Other analyses that would not be significantly affected would continue to rely upon the current licensing basis methodology.	After the sentence: "The NRC staff expects licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately." Add the sentence: "Since the existing licensing basis methodology remains valid, a licensee may use the guidance in DG-1113 on an event basis selectively. It is not necessary that all the existing event based licensing analyses be updated for each application of the guide."
6	5	C.1.3.2	The last sentence of this paragraph infers that a license amendment request is necessary for each reanalysis. This could become burdensome to the NRC staff. Certain reanalysis changes should be authorized via this regulatory guide without submittal to and review by the NRC if required by 10CFR50.59. Two examples are: • Implementation of ICRP-30 dose conversion factors (Position 4.1), and • The Control Room (Dose) Acceptance Criteria (Position 4.5)	Add the following sentence: "Licensees may implement the ICRP-30 dose conversion factors (Position 4.1) and the Control Room (Dose) Acceptance Criteria (Position 4.5) without a submittal to the NRC if required by 10CFR50.59."
7	5	C.1.3.2	The last sentence infers that reanalysis should be performed for all areas in Section C.1.3.1, including areas unrelated to control room habitability such as Equipment Qualification and Accident Monitoring Instrumentation. This is inappropriate. Provide for selective implementation of this regulatory guide as discussed in our comment on Section C.1.3.1, first paragraph. Delete this sentence, since implementation of this criterion will be burdensome to both the NRC staff and licensees without commensurate improvements in safety.	Delete the last sentence of Paragraph C.1.3.2

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8	10	C.2.8, Eq. 11	The multiplicative constant (352) in the equation $352(V_k^{0.338})$ is given to apply when V_k is given in units of cubic meters. However, compartment volume is usually calculated in cubic feet. This could result in conversion errors. Include the constant (1173) to give $1173/(V_k^{0.338})$, when the free volume of the compartment is given in units of cubic feet.	Adopt the formulation provided in the comment.
9	11	C.3.1.1, Footnote 4	Footnote 4 only addresses the reactor head drop accident. However, there are other accidents where multiple assemblies are postulated to be damaged. Therefore, the footnote should be re-written to encompass theses other accidents.	Rewrite Footnote 4 to read: "For accidents which involve several assemblies, it is appropriate to use the core average inventory."
10	11	C.3.2, Footnote 5	This footnote refers to the release fractions in Table 1, where all noble gases are presumed released and 50% of the iodines are assumed to be instantaneously released. Data and evaluation for MOX fuel support a conclusion that these models and values are also conservative for MOX fuel application using this licensing approach. Since conservative release fraction values and timing assumptions are already specified for this application, the conditional statement regarding MOX fuel presented in the last sentence should be deleted.	Delete the last sentence of Footnote 5.
11	12	C.3.2, last	The definition of fuel melt and calculation of material release should be clearly specified, and the bases should be justified for non-LOCA events. Major fuel failure consequences are postulated for the DBA LBLOCA scenario and that is the condition for which Table 1 applies. Table 1 should be labeled as such. Table 2 is not for postulated "fuel melt". It is applicable to postulate cladding damage that results in a breach of cladding (e.g., using the DNB criterion, which in itself is generally very conservative). If Table 1 is the proper citation in the last sentence, then it creates a situation that introduces unnecessary conservatism, because for non-LOCA events, fuel rods that are calculated to experience incipient fuel melting temperatures should be conservatively assigned as fuel with cladding failure. Fission gas release fractions of 1.0 are normally assigned to fuel material that has melted, not to the surrounding fuel material at lower temperatures. For non-LOCA events the extent of fuel melting within a fuel rod should be considered in calculating isotopic release fractions from these failed fuel rods.	Re-title Table 1 for application to the DBA LOCA. Replace "fuel melt" with "cladding damage" in Table 2 references. Define "fuel damage" as terminology for a general characterization of fuel material effects; e.g., (fuel melting, fuel thermal induced changes), cladding perforation, or both.

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12	12	C.3.2 Footnote 7	Footnote 7, which states that "fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis", should be applicable to both low enriched uranium fuel and MOX fuel analyses, presuming that the fuel design is approved by the NRC Staff and the provisions of power history modeling are followed as prescribed.	Revise the sentence to read: "As an alternative, fission gas release calculations performed for either low enriched uranium fuel or MOX fuel using NRC-approved methodologies may be considered on a case-by-case basis."
13	14 15	C.4.2.1	This section lists potential "sources of radiation that will cause exposure to control room personnel .... typically will include." Of these sources, only the first source, "Contamination of the control room envelope atmosphere by the intake or infiltration of the radioactive material in the radioactive plume released from the facility," is typically included in rigorous analyses of radiological consequences of design basis accidents. The remaining sources have been addressed in the past by design practice, qualitative engineering evaluations, or simple engineering estimates. These approaches are supported by descriptions and expectations provided in the Standard Review Plan (SRP), especially SRP 6.4, Sections I.4, I.5, and III.3 through III.7. Requiring these additional sources to be included in analysis of radiological consequences of design basis accidents is a significant departure from these past practices approved by the NRC staff. In addition, models necessary to analyze these additional sources are not generally compatible with either the models presented in Section C.2 or similar models. Source 2: "Contamination of the control room envelope by the intake or infiltration of airborne radioactive material from areas and structure adjacent to the control room envelope" is intractable. Source terms in these areas would be due to transport from other rooms (e.g., Mechanical or Electrical Penetration Rooms, Rooms with ECCS equipment), where radioactivity has to diffuse past closed doors and piping penetrations to arrive at the rooms adjacent to the control room envelope, or from the outside. Detailed internal transport modeling is not compatible with the model of draft Regulatory Position C.2. If the radioactivity is transported to these rooms from the outside, then this pathway may be represented with a control room χ/Q either for unfiltered in-leakage or, if conservative or appropriate, filtered airflow. In the past, qualitative engineering evaluations alone have addressed this source term (SRP Sections I.4, III.5.a).	Delete this guidance from DG-1113 and include a reference in DG-1114 ("Control Room Habitability at Light-Water Nuclear Power Reactors") to the modeling provided in DG-1113 for analysis of, "Contamination of the control room envelope atmosphere by the intake or infiltration of the radioactive material in the radioactive plume released from the facility." This may be appropriate either as an additional bulleted item in Section 1.1 or, alternatively, under Section 2.3.2. Also provide in DG-1114 direction to evaluate the remainder of the applicable sources in a manner consistent with previous regulatory guidance and NRC Staff endorsements and/or approvals as referenced in the SRP.

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			Source 3: The position pertaining to "Radiation shine from the external radioactive plume released from the facility" is intractable also. No positions concerning modeling features such as the spatial distribution of the specific activity or effects of dispersion are given. Shielding provided by Seismic Category I control room boundary and enclosing and adjacent structures will make this source contribution to control room operator dose negligible. In the past, simple engineering estimates have been acceptable to demonstrate this. (SRP Sections I.4, I.5, III.5.a, III.6, III.6.a, III.6.b, and III.6.c). Source 4: The position concerning "Radiation shine from radioactive material in the reactor containment" will create unnecessary conservatism. The thickness of the Seismic Category I reactor building makes this source contribution negligible. In the past, qualitative engineering evaluations or simple engineering estimates have acceptable to demonstrate this. (SRP Sections I.5, III.5.a, III.6, III.6.a, III.6.b, and III.6.c). Source 5: The position concerning "Radiation shine from radioactive materials in systems and components external to the control room envelope…." is unnecessary. Dispersion of the radioactivity before it is taken into these components, combined with shielding provided by area structures and the control room walls should results in negligible dose from this source. (SRP Sections I.4, III.5.a, III.6, III.6.a, III.6.b, and III.6.c). One potential source not included in this section is radiation shine from ESF intersystem leakage material that accumulates in the Refueling Water Storage Tank (Generic Letter 91-56). This source may be a minor contributor, based on distance from the control room and shielding provided by the control room walls. However, it is likely a more significant contributor than sources (3) and (4) above.	

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14	17	C.4.5	Section C.4.5 has increased the thyroid dose guideline from 30 to 50 rem, but leaves the skin dose limit at 30 rem, and doesn't retain the alternate guideline of 75 rem skin allowed in the SRP Section 6.4 for licensees who provide protective clothing and eye protection. The alternate limit for protective clothing and eye protection from the Standard Review Plan should be retained, especially if part of the Licensing Bases for a facility. Additionally the skin guideline without protective clothing or equipment should be increased from 30 rem to 50 rem consistent with the thyroid guideline. 10CFR20.1201limits skin dose to 50 rem annually. A weighting factor of 0.06 is specified for the skin in 10 CFR20.1003, so 83.5 rem to the skin represents an equivalent whole body dose of 5 rem. The guideline for dose to the eye without protection should be 30 rem. Therefore, the same justification exists for an increase in the skin dose guidelines as exists for the increase in the thyroid dose guidelines from 30 rem to 50 rem. This will prevent arbitrarily making the skin dose the limiting consideration due to an inconsistent consideration of the GDC-19 criteria of 5 rem whole body, or its equivalent to any part of the body. The SRP section 6.4 guideline values of 30 rem for the skin and thyroid were derived from previous annual occupational dose limits for the skin and thyroid in ICRP Publication 2. The new guideline values for the skin and thyroid should have a consistent basis. This is further justified when considering the low weighting factor of 0.01 recommended for the skin in ICRP-60.	Revise section C.4.5 to read: **4.5 Control Room Acceptance Criteria** The following guidelines may be used in lieu of those provided in SRP 6.4 (Ref. 14) when showing compliance with the dose guidelines in GDC-19 of Appendix A to 10 CFR Part 50. The following guidelines relax the thyroid and skin acceptance criteria from that given in SRP 6.4. This relaxation from 30 to 50 rem is based on a change to 0.03 in the thyroid organ dose weighting factor given in 10 CFR 20.1003, and 0.06 in the skin dose weighting factor. Although this change gives an equivalent thyroid dose of 167 rem-thyroid and 83 rem-skin, 10 CFR 20.1201 limits organ and skin dose to 50 rem annually. The release duration is specified in Table 4. The exposure period is 30 days for all accidents. The criterion in GDC-19 applies to all accidents. Whole body 5 rem Thyroid 50 rem (See Comment 15) Skin 50 rem* Eye 30 rem* *Credit for the beta radiation shielding afforded by special protective clothing and eye protection is allowed if the applicant commits to their use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose is not to exceed 75 rem. The skin and thyroid dose levels are to be used only for judging the acceptability of the design provisions for protecting control room operators under postulated design basis accident conditions. They are not to be interpreted as acceptable emergency doses.
15	17	C.4.5	This section states that the thyroid dose, equivalent to the GDC 19 limit of 5 rem, is 167 rem with a thyroid organ weighting factor of 0.03 (or, alternatively, 100 rem with the thyroid organ weighting factor set to 0.05), but cites 10 CFR 20.1201 for limiting the thyroid dose to 50 rem.	Either revise C.4.5 to allow a control room thyroid radiation dose of 100 rem, or revise R.G. 1.183 to give positions on organ radiation doses based on ICRP-30 and ICRP-60.

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			The limit of 50 rem is based on ICRP-30 and confirmed in ICRP-60. However, DG-1113 did not take corresponding positions in R.G. 1.183 for thyroid radiation dose in particular and organ radiation doses in general. Rather, it took the position that these limits could be ignored based on low risk of organ doses. DG-1113 concluded that adequate protection could be ensured by setting a limit on a TEDE (implicitly acknowledging that organ function was not actually covered except by the argument on risk). This is an inconsistency that cannot be explained by the "Alternative Source Terms."	Document the justification for the approach taken.
16	17	C.4.6, 2nd	No compelling reason exists for a licensee to revise its basic design criteria. Therefore, a sentence should be added to affirm that these current licensing basis commitments remain acceptable. DG-1113 has defined the scope of its application to those general design criteria (GDC) specified in Section A. Therefore, the regulatory guide does not apply to plants that do not have the specified GDCs as part of its licensing basis. The regulatory guide should be modified to define the scope of plants designed to criteria other than the GDCs or the regulatory guide should acknowledge that it is only applicable to plants designed to the GDCs listed in Section A.	Revise the sentence to read: "Although these commitments may be different from GDC-19, the continued use of these current licensing bases remain acceptable" Modify Section A to include the scope of plants designed to criteria other than the GDCs or indicate that the guide is only applicable to plants designed to the GDCs listed in Section A.
17	18	C.5.1.2	The design basis and current licensing bases for some plants may allow credit for some non safety-related equipment in the radiological analyses of DBAs at these plants. Two examples are given. • The calculation of radiation doses for the "Break of a Small Line Carrying Reactor Coolant Outside of Containment (SRP 15.6.2, "small line break") by some licensees includes credit implicitly taken for non safety-related instruments to detect the small line break. If, for example, the break is in the letdown line, credit may be taken for radiation monitors in the area or level instrumentation for the Volume Control Tank of the Chemical and Volume Control System, all of which may be non safety-related. • Some radiological consequence evaluations for DB Steam Generator Tube Rupture (SGTR) take the position that two	Rewrite this section to allow for continuation of credit for non safety-related equipment as permitted in various plant license bases.

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			systems or components should be available for each action required to limit the activity releases following the DB SGTR, but that one of the two systems or components may be non safety-related. This section should be written so as to allow the continuation of credit for non safety-related equipment as allowed in the current license bases for individual nuclear plants.	
18	18 E-2 F-3 G-2	C.5.1.2 Appendices E.2.4, F.2.5, G.2.4	The following sentence is inconsistent with licensing basis of some plants, where loss of offsite power is not considered or assumed to occur coincident with the analyzed event: "Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences." This sentence should be changed to reflect the current licensing basis for plants regarding the assumptions and occurrence of loss of offsite power. This comment is applicable to Appendices E, F, and G. The same changes should be made to these appendices.	In Section C and Appendices E, F and G change the sentence to read: "Assumptions regarding the occurrence and timing of a loss of offsite power should be selected based on current licensing basis requirements"
19	18	C.5.1.3	Once the course of an accident sequence is set, it does not change. Therefore, to modify a sequence to continually present a "worse case" set of conditions in the evaluation is unnecessarily conservative. However, accident sequence and plant system response assumptions should remain consistent for the entire set or series of analyses that provide input to and comprise all elements of the radiological dose analyses. The guidance as written will present an intractable, unbounded licensing analysis task, seeking sets of conditions that may provide more conservative results. One set of consistent analysis assumptions should be used throughout the analysis of a given event sequence.	Prior to the sentence: "Sensitivity analyses may …", insert the following sentence: "Consistent modeling of performance of engineered safety features' operation should be used for the course of an accident sequence and for the entire set or series of analyses that provide input to and comprise all elements of the radiological dose analyses." In the next sentence replace "appropriate values to use" with "the consistent set of accident sequence and equipment performance, modeling assumptions, and values that will result in an appropriately conservative licensing basis evaluation method".
20	19	C.5.2 1st	This paragraph limits application of DG-1113 to accidents involving damage only to irradiated fuel. No guidance is provided for accidents with source terms that are not associated with fuel or cladding damage.	Delete the second, third and fourth sentences from the first paragraph of C.5.2 and replace them with the following sentence:

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			This apparently is not the intent. Positions are taken pertaining to the pre-accident iodine spike and accident initiated iodine spike in the absence of any fuel damage for the DB SGTR and the DB Main Steam Line Break (MSLB). In addition, limits on offsite radiation doses for these DBAs with these iodine spikes are prescribed in Table 4. This contradicts the statements in the 1st paragraph of Section C.5.2. Several other Appendices pose the possibility that no fuel damage would occur during the postulated event.	"Licensees should review their license basis document for guidance pertaining to the analysis of radiological consequences of other design basis accidents."
21	20	C.5.3	The last sentence states: "All changes in χ/Q analysis methodology should be reviewed by the NRC staff." DG-1113 should establish a minimal threshold for when the NRC staff does not need to see a change. An example would be inclusion of more recent weather data, while using the same method (for instance, Murphy-Campe). Requiring that all changes to χ/Q methodology be submitted to the NRC is in conflict with 10 CFR 50.59, since the 50.59 process permits NRC accepted methods to be used by other plants without NRC prior approval. Furthermore, Section 4.1.1 of NEI 96-07, *Applicability to Licensee Activities,* which is endorsed by the NRC staff in RG 1.187, states that: "10 CFR 50.59 is applicable [...] to changes to the facility or procedures as described in the Updated Final Safety Analysis Report, including changes made in response to new requirements or generic communications". DG-1111 (p.4) states that "holders of operating licenses may continue to use χ/Q values determined with methodologies previously approved by the NRC staff and documented in the facility's FSAR to the extent that these values are appropriate for the application for which they are being used. Licensees may also continue to use the licensing basis methodology for determining χ/Q values for newly identified source-receptor combinations or re-generating the approved χ/Q values using more recently collected meteorological data sets." DG-1113 states that: "Atmospheric dispersion values (χ/Q) for the EAB,	Remove the sentence: "All changes in χ/Q analysis methodology should be reviewed by the NRC staff". Modify this section to be consistent with DG-1111.

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			the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide provided such values remain relevant to the particular accident, its release points, and receptor location. … References 18 (Murphy-Campe) and 26 (ARCON96) should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor differences." These two positions are not consistent. It is recommended that consistency be established by adopting the position in DG-1111.	
22	20	D, 2nd	The second sentence of the second paragraph reads: "Except in those cases in which an applicant or licensee proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods to be described in the final guide reflecting public comments will be used in the evaluation of submittals in connection with radiological consequences at nuclear power reactors." This statement exceeds the guidance presented on the cover of every regulatory guide issued by the NRC staff. In the standard statement is: "Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluation specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutions for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission." Furthermore the 2nd sentence of the 2nd paragraph in Section D establishes DG-1113 as a de-facto regulation, rather than one acceptable method to satisfy the regulations. This sentence infers that final version of DG-1113 will be used as a metric for comparing all other acceptable methods in lieu of the regulations. Furthermore, the implementation section does not address how the regulatory guide is to be used in conjunction with the licensee's existing licensing basis.	Replace this sentence with: "Regulatory Guides are issued to describe and make available to the public methods acceptable to the NRC staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluation specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutions for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission."

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			Section B acknowledges that the guidance contained in DG-1113 … "will supercede corresponding radiological analysis assumptions provided in other guides when used in conjunction with guidance... DG-1114". Then it is stated "The affected guides will not be withdrawn as they may still be used at the options of the licensees." However, the current DG-1113 Section D statement imposes guidance that constitutes backfitting per 10 CFR 50.109 because Section D established this regulatory guide as a metric for comparison of all other acceptable methods. Paragraph (a)(1) of the backfitting rule states: "(a)(1) Backfitting is defined as the modification … procedures … which may result from a new or amended … regulatory staff position interpreting the Commission rules that is either new or different from a previous staff position …" This becomes particularly evident since DG-1113 is implemented through conformance to DG-1114, which states "The application of this regulatory guide may involve … change to the licensing basis of the facility." Since regulatory guides are not substitutions for regulations, and compliance with them is not required, Section D should be revised to reflect the official text placed on the cover of each NRC staff issued regulatory guide.	
23	A-4	Appendix A.4.1	The "deterministic approach" outlined for treatment of ESF leakage is not realistic and is overly conservative. The staff should consider endorsing a more reasonable, but conservative position as follows: Two DB LOCA scenarios would be analyzed. The first scenario would assume that all released core iodine inventory (50%) is in the sump. Then all radiation doses would be calculated as if they were associated with the ESF component leakage release pathway only. The second scenario would limit the sump iodine inventory to a value calculated by applying the spray decontamination factor (spray DF). Radiation doses for both containment leakage and ESF component leakage release pathways would be computed for this scenario. The higher radiation dose would be taken as the limiting radiological consequences of the DB LOCA.	Revise the first sentence: "To incorporate the dose related to the ESF system leakage in a conservative manner, two DB LOCA scenarios should be analyzed. The first scenario assumes that all released core iodine inventory (50%, based on the maximum power level) is mixed instantaneously and homogeneously in the primary containment sump water (in PWRs) or the suppression pool (in BWRs) at the start of the accident. Then all radiation doses are calculated as if they were associated with the ESF component leakage release pathway only. The second scenario would limit the sump iodine inventory to a value calculated by applying the spray

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			This method would be an improvement over that prescribed in the "deterministic approach", where all released core iodine inventory is assumed to be transported both to the sump release pathway and to the containment atmosphere release pathway, so that some core iodine inventory release to the environment will be counted twice. The option to develop "suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water" should be retained.	decontamination factor (spray DF). Radiation doses for both containment leakage and ESF component leakage release pathways would be computed for this scenario. The higher radiation dose would be taken as the limiting radiological consequences of the DB LOCA."
24	B-1	Appendix B.2, 1st	The values provided in this paragraph do not form a consistent set of calculation results. This may lead to confusion and misuse of this guidance. In fact, the inconsistent factors and composition percentages would result in more organic iodines above the pool than had actually been released from the fuel cladding gap in this accident (suggesting that the pool is adding organic iodines). The only specific pool DFs compatible with the regulatory positions (effective DF = 200 and composition fractions for iodine in the fuel pin gaps and leaving the pool) are 448.9 for elemental iodine and 0.9 for organic iodine. An organic iodine DF of 0.9 is equivalent to postulating some elemental iodine converting to organic iodine compounds before it leaves the pool. The recommended formulation assigns the specific iodine DFs to 500 and 1, and assigns the composition fractions for iodine in the gap to 99.75% elemental and 0.25% organic. This yields an overall effective decontamination factor of 222. The mixture of species released from the pool matches that of the draft guidance, 44% elemental and 56% organic species.	Revise the first portion of this paragraph to read: "If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors (DF) for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 222. This difference in decontamination factors for elemental (99.75%) and organic iodine (0.25%) species results in the iodine above the water being composed of 44% elemental and 56% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B 1)."
25	B-2, B-3	Appendix B.4.1 B.5.3	No specific guidance on the time dependent profile for release is provided. Since the release is stated to be from a building (containment or fuel building), the underlying mechanism appears to be holdup and release from a constant volume.	Add the following sentence to 4.1 and 5.3: "The release rate is a function of the plant configuration, but is generally assumed to be a linear or exponential function over this time period."
26	C-1 H-1	Appendix C.1 Appendix H.1	The last sentence describes fission product release from fuel that has reached fuel melting temperature. The term: "the fraction of the fuel" is used twice in the sentence and should have two distinct meanings. These are "fraction of the fuel rods that have failed or breached cladding" or "fraction of the fuel material that has melted in a rod."	Rewrite the sentence as follows: "The release attributed to fuel melting should be based upon the fraction of the fuel rods that have breached cladding due to incipient fuel melting, combined with volumetric fraction of fuel material that has melted wherein it is assumed that 100% of

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			The sentence should be revised to clarify that the distinction between these meanings and so that licensees will provide a better prediction of fission products released from melted fuel. These comments, with a modification on the assumed percentages of iodine that is released, also apply to Appendix H as noted.	the noble gases and 50% of the iodines are released to the reactor coolant." Add the sentence: "The option of using other criteria to determine fuel cladding failure, such as fuel centerline or planar average energy deposition, is also acceptable."
27	E-2, E-3, F-3	Appendix E.2.5 E.2.6	The descriptions of leakage flashing are confusing and the modeling expectations regarding flashing are unclear. The discussion from EPRI TR-107621, Revision1, Appendix K (Radiological Assessment Guidelines) – and reference to it – would clarify the most appropriate approach: "Note that the iodine released due to flashed break flow is not a consequence of uncovery of the tube bundle, as this flashed fraction is released even when the break site is below the top of the water level (Here we are referring to large leak rates such as with steam generator tube rupture flow rate. Small leaks such as would be the case with technical specification limits on primary to secondary leak rate will mix with boiler water if and only if the tube bundle is covered, that is, credit may be assumed for small leaks which are covered by water). Existence of water over the top of the break site indicates that credit may be taken for partial scrubbing of the iodine contained in the flashed flow. Models for scrubbing credit may be found in work that has been performed by Postma and Tam, although the specific applicability of this credit should be evaluated by each licensee prior to use." The transport model should recognize: 1. Flashed flow occurs even with the tube bundle being covered. 2. The existence of water over the tube bundle means only that credit may be taken for scrubbing of the flashed flow. The discussion on primary bypass indicates that during periods of uncovery, small leaks entirely escape (i.e., flashed flow + primary bypass = unity). 3. Small leaks will mix with boiler water "if and only if the tube bundle is covered." In summary, during periods when the tube bundle is covered, large leaks such as with the SGTR flash, but do not entrain primary bypass, and during this period, credit may be taken for partial scrubbing of the	Sections E2.5.1 through E2.5.4 should be reorganized and amended as follows: 2.5.1 A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant. 2.5.2 During periods of total submergence of the tubes in the affected steam generator, the primary-to-secondary break flow that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models of NUREG-0409, "Iodine behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-1). 2.5.3 When the steam generator tubes are covered, the primary-to-secondary leakage that does not immediately flash is assumed to mix with the bulk water. Leakage that has been apportioned to the affected steam generator, as described in Section 2.1, will mix with the bulk water, so that there is only steaming, and no flashing or primary bypass. Likewise, during periods of total submergence of the tubes in the unaffected steam generators used for plant cooldown, apportioned leakage will mix with the bulk water. 2.5.4 The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient.[3] A partition coefficient for iodine of 100 may be

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			flashed flow, depending upon the applicability of the scrubbing model. During periods when the tube bundle is uncovered and still for large leaks, primary bypass is entrained within the flashed flow. For small leaks during periods when the tube bundle is covered, the small leak will mix with the bulk water, so that there is only steaming, and no flashing or primary bypass. For small leaks with tube bundle uncovery, flashing + primary bypass = unity. Note that in the Recommended Revision, Section 2.5.4 has been revised to delete the sentence discussing particulates as discussed in the following comment.	assumed. 2.5.5 Under conditions of tube uncovery, the transport model parameters should be evaluated to include consideration of both the flashed vapor and the primary bypass that is entrained within the flashed flow. This would apply to both the break flow and leakage flow in the affected steam generator and to the leakage flow in the unaffected steam generators if either region experiences conditions of tube uncovery.
28	H-1	Appendix H.1, 1st	The general comments provided for Appendix C also apply. The release fractions specified here to calculate radioactive isotopes available for release from the containment are different. The calculation framework, either in terms of temperature or enthalpy deposition should be similar. Fuel rods that experience incipient centerline melt are conservatively assumed to be failed. These rods are then subject to the release fraction associated with the release of inventory from the fuel-cladding gap. Additionally, the fraction of the fuel material that is calculated to have melted in the rod is subject to the radioactive isotope release fractions specified for melted fuel material.	More specific guidance related to the calculation of the release fractions of radioactive isotopes from the core should be provided. The licensee should have the option to evaluate the core damage consequences due to fuel failures attributed to fuel melting in fuel rods in more detail. For clarity the third and fourth sentences can be rewritten as: "The release attributed to fuel melting is based on the fraction of the fuel material that reaches or exceeds fuel melting temperature. For this fuel material fraction, the assumption is that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines contained in that fuel material fraction are released to the reactor coolant."
29	H-2	Appendix H.2.2	This section establishes the containment leak rate assumption for the Rod Ejection accident. DG-1113 identifies the leak rate as equivalent to a LB LOCA, even though the calculated containment response for a rod ejection is less severe. Use of a more realistic containment leak rate for a rod ejection accident is appropriate because the transient and peak containment pressure, and timing of the containment response will be much less severe than a LB LOCA, which assumes a double-ended pipe break of a larger pipe.	Provide a footnote to 2.2 stating: "Licensee may propose modification of the containment release rate on a plant specific basis, since, the containment leak rate associated with the PWR rod ejection accident is less severe than that associated with a LB LOCA."

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			EDITORIAL COMMENT	
30		C.4.1 C.4.2 C.4.4	Table 4 lists EAB and LPZ dose criteria that would be better placed to in Section 4.1 on offsite dose consequences or Section C.4.4 on offsite acceptance criteria. Currently, it is located in the middle of section C.4.2 on CR dose consequences.	Relocate Table 4 to Section 4.1