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March 31, 2000

Ms. Annette Vietti-Cook
Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Rules and Directives Branch

SUBJECT: Draft Regulatory Guide DG-1081, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, (64 *Fed. Reg.* 71990)
Request for Comments

PROJECT NUMBER: 689

Enclosed are the Nuclear Energy Institute's (NEI)[1] comments on draft Regulatory Guide DG-1081, issued for public comment on December 23, 1999. Also included are comments on the associated Standard Review Plan section 15.0.1.

Enclosure 1 provides a response to the specific questions regarding scope of implementation and reanalyses identified by the NRC staff in Section IV of the *Federal Register* notice. Enclosures 2 and 3 provide technical and editorial comments, respectively, on DG-1081. Enclosure 4 provides comments on the proposed Standard Review Plan section.

We would welcome the opportunity to meet again with the NRC staff to explain the industry comments, if deemed helpful.

[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.

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Please direct any clarification requests or questions to Kurt Cozens at (202) 739-8085 or koc@nei.org.

Sincerely,

David J. Modeen

KOC/
Enclosures

c: Mr. Richard J. Barrett, U.S. Nuclear Regulatory Commission
Mr. Mark F. Reinhart, U.S. Nuclear Regulatory Commission
Mr. Steve F. LaVie, U.S. Nuclear Regulatory Commission

Comments to Federal Register Questions

The December 23, 1999 issue of the *Federal Register* (page 71990 through 72002) noticed the availability of draft Regulatory Guide DG-1081 for public comment. In addition, Section IV of the *Federal Register* notice requested a response to a series of questions regarding the scope of implementation and re-analyses of alternative source term. The following is the industry response to these questions.

Scope of Implementation

Question 1.A *Does the proposed guidance provide the desired flexibility while providing reasonable assurance that a clear, consistent, and logical design basis will be maintained?*

Yes. DG-1081, in permitting either full or selective implementation, provides some flexibility while maintaining the objective of a clear, consistent and logical design basis. Suggestions for additional flexibility are included in the industry comments provided in Enclosure 2.

Question 1.B *Is there a less complex alternative approach that would provide the desired flexibility while maintaining a clear, consistent and logical design basis?*

No. The Commission's direction to use the TEDE criteria as the basis for determining acceptable use of an alternative source term if it were to be implemented in a comprehensive manner by a licensee, resulted in the need for development of a rule. The use of a regulatory guide to provide NRC staff guidance on at least one acceptable approach to implement the rule requirements is appropriate. The industry appreciates the effort to outline acceptable means for the implementation of the AST and has no suggestion for a less complex approach.

Question 1.C *Should the Commission allow licensees that have received approval for a selective implementation to extend the AST and TEDE criteria to other design basis applications (that do not involve reanalysis of the DBA LOCA) under 10 CFR 50.59 rather than under 10 CFR 50.67 as currently proposed?*

Yes. Revised 10 CFR 50.59, approved by the Commission in June 1999[1], contains an explicit criterion that is directly applicable and well suited to controlling expanded use of the AST by licensees previously approved for selective implementation. Indeed, the rule criterion and associated guidance (summarized below) has been developed precisely for the purpose of controlling key analytical

[1]. The revised 10 CFR 50.59 rule will become effective 90 days after associated regulatory guidance is approved (approximately February 1, 2001).

methodologies such as for source term and LOCA, that underlie UFSAR accident analyses.

In light of the explicit, effective control provided by 10 CFR 50.59, the 10 CFR 50.67 requirement that a license amendment be obtained for each selective application of the AST after the first is overly burdensome and unwarranted.

Discussion

The existing 10 CFR 50.59 does not explicitly control licensee changes to safety analysis methodology such as the methodology for defining the source term on which accident analyses are based. The NRC decision in § 50.67 to require use of the AST to be approved by license amendment reflects, at least in part, this lack of explicit control. However, revised § 50.59, which was completed in parallel with the § 50.67 rulemaking,

(1) considers methods of evaluation to be part of the "facility as describe in the UFSAR," and thus within the scope of the revised rule, and

(2) criterion c(2)(viii) requires prior NRC approval if a change would "result in a departure from a method of evaluation described in the FSAR (as updated) used to establish the design bases or in the safety analyses."

Revised § 50.59 defines departure from a method of evaluation as described in the FSAR (as updated) as either: (i) changing any of the elements of the method described in the FSAR (as updated) unless the results of the analysis are conservative or essentially the same; or (ii) changing from a method described in the FSAR to another method unless that method has been approved by NRC for the intended application.

The NRC is in the process of endorsing proposed industry guidance for implementing the revised rule. A *Federal Register* notice is expected to be published in April 2000 seeking public comment on Draft Regulatory Guide 1095 endorsing NEI 96-07, Revision 1, *Guidelines for 10 CFR 50.59 Evaluations.*

Section 4.3.8.2 of NEI 96-07, Revision 1, provides guidance for determining whether an alternate methodology is "approved by the NRC for the intended application." A licensee may adopt an alternative method, without prior NRC approval, provided

(1) the method has been specifically approved by the NRC for the intended plant/application, or

(2) the licensee is qualified per GL 83-11, Supplement 1, to perform safety analyses and determines that the change is technically appropriate, consistent with conditions and limitations applicable to the intended use of the method, and consistent with the licensing/design bases for the plant.

Recommendation

In regulatory guidance for implementing § 50.67, the NRC should clarify that licensees who have received approval for selective implementation of AST and are qualified per GL 83-11, Supplement 1, may determine consistent with § 50.59 and NEI 96-07, Revision 1, whether or not additional selective implementations of the AST (that do not involve reanalysis of the DBA LOCA) require prior NRC approval. Specifically, such licensees should be permitted to

(1) evaluate selective AST applications approved for other plants to determine if they are appropriate for application at their own plant, and

(2) extend the AST and TEDE criteria to additional technically appropriate applications at their plant

There are other forms of selective implementation that also should not be constrained by § 50.67(b). That is, those applications of alternative source term insights that are beyond the intent of § 50.67 to control, i.e., direct replacement of the source term used in design basis radiological consequence analyses. For example, consider the activity identified in a BWR Owners Group topical report—that we understand the NRC staff has approved—to define the timing for BWR fuel gap releases. Alternative source term insights reflected in this report have been used by licensees with the TID-14844 source term and the whole body/thyroid dose criteria to implement changes in allowed closure time limits for key containment isolation valves. Application of alternative source term insights such as this are not subject to the new rule and its associated requirements. Consequently, licensees should use § 50.59 as discussed above to evaluate and implement such applications of alternative source term insights.

Finally, the rule language promulgated in §50.67 was drafted at a time when the final form of the revised § 50.59 rule and guidance was not known. Given the effective control of methodology provided by the revised § 50.59 rule, the requirements in §50.67 are more restrictive than they need to be. After the industry and NRC gain some experience in applying the alternative source term rule and implementing guidance, industry will evaluate if a revision to §50.67 would be beneficial and engage the NRC, as appropriate.

Questions 2.A & B *What other combinations of AST characteristics are technically consistent? What plant modifications might be based on these combinations?*

These questions do not have a simple answer that is both concise and complete. However, Comment # 6 of Enclosure 2 provides two specific examples where selective implementation should be permitted. In assessing the totality of the industry comments on the draft regulatory guide, as well as previous NRC staff reviews of licensee-specific proposals, there are surely other combinations or permutations that warrant consideration as selective implementation of an AST.

SCOPE OF RE-ANALYSES

Question 1.A *Is the proposed guidance on the scope of re-analyses technically appropriate and clear? How could it be improved?*

The draft regulatory guide is comprehensive and contains an appropriate level of detail. Nonetheless, as the industry comments provided in Enclosure 2 indicate, there are several areas of the document that warrant clarification or reconsideration of the proposed guidance. These comments are the result of a thorough review of the draft guidance by a multi-disciplinary task force of industry representatives. The comments should be given careful consideration and thoughtful response by the NRC in order to strengthen the value and usefulness of the regulatory guide.

Question 1.B *The guidance allows licensees to disposition certain impacts of an AST on the basis of the NRC staff's rebaselining study. Does this study or other documents provide a sufficient basis for the Commission to generically disposition these impacts?*

Yes. The NRC baseline studies and the pilot plant applications, either completed or in progress, provide an excellent basis for the Commission to disposition generically the impact of using an alternative source term. It is not clear that all of the insights from the pilot plant reviews are reflected in the current draft of DG-1081. The NRC staff should cross-reference the technical positions taken in the pilot plant reviews to assure that the positions accepted by the NRC in the pilot plant evaluations are also reflected to be acceptable in the final regulatory guide.

Question 2.A *Should the Commission allow licensees to continue to use the prior source term and dose criteria for these analyses [those using prior source terms and methodologies] and not require that they be updated on subsequent revision?*

Yes. Industry concurs with the NRC that the baseline studies provide an adequate basis to conclude that use of the DG-1081 alternative source term is appropriately bounded by prior source term analyses.

Question 2.B *If the analyses are not updated, how will licensees assure that the earlier conclusion that the analyses are limiting remains valid following subsequent revisions?*

As with all commitments, the burden is on the licensee implementing the alternative source term to assure that compliance with all regulatory requirements is maintained. No additional regulatory guidance is necessary to provide that assurance.

Question 3.A *Is there information that should be considered by the Commission in resolving this generic issue?*

The NRC has previously evaluated accident radiation dose for equipment qualification purposes. NUREG/CR-5313 (SAND-88-3330), *Equipment Qualification Risk Scoping Study*, concluded that: (1) the importance of the accident radiation dose is overemphasized, (2) that equipment qualification issues associated with long term accident equipment operability are not risk significant, and (3) equipment qualification should focus on ensuring equipment operability for the first few days of the accident exposure, as illustrated by plant risk assessments.

Industry understands that the above is the basis for the current NRC position that there is no safety concern relative to equipment qualification for plants licensed to TID-14844, even though, based on 30 years of research which evolved into the AST, it is estimated that after 30 days, the integrated dose to safety equipment exposed to sump water may be slightly higher than that predicted by the TID methodology.

Industry is unaware of any additional information that would assist the NRC in dispositioning the generic safety issue raised by the postulated increase in cesium concentrations.

Question 3.B *If the Commission should conclude that there is safety significance but that the costs are not justified on a generic basis, should licensees who are voluntarily proposing to amend their design basis to use an AST be required to address the impact of the increased cesium concentration?*

No. Any NRC staff decision that has the net effect of imposing new requirements must be supported by a regulatory analysis conducted in accordance with the provisions of the backfitting rule, §50.109. Voluntary licensee actions relative to use of the alternative source term should not be predicated on a licensee commitment to address other safety significant insights that do meet a backfitting test. It would be poor regulatory decision making policy to pursue such regulatory action in the form of a *quid pro quo*.

As described in Comments # 15 and 19 of Enclosure 2, industry recommends a more consistent definition or standardization of the time period used in calculating accident doses. Without such a standard, generic resolution of the cesium concentration issue is more difficult. The periods we suggest are consistent with the period used in the NRC re-baseline studies. We note that those studies demonstrated that the TID source term doses are conservative.

Question 3.C *If a licensee proposes a change in plant configuration that would result in an increase in the integrated dose for one or more components and this licensee is also proposing, or has already implemented an AST, should the re-analysis of the integrated dose be based on that AST or on the prior TID14844 source term?*

The provisions of paragraph 1.2.1 to DG-1081 direct licensees to establish the alternative source term as the licensing basis if full implementation relative to radiological dose analyses is pursued. That section makes it clear that the NRC intended that for any future radiological dose analyses the TEDE dose criteria is the figure of merit as opposed to thyroid or whole body dose criteria. These provisions are silent as to what is intended relative to radiological equipment qualification (integrated dose) evaluations for structures, systems and components.

Based on the current knowledge, it appears to be an unjustified regulatory burden to require licensees to redo equipment integrated dose evaluations using the alternative source term. The NRC staff rebaselining study concluded that the TID-14844 source term evaluations are conservative. Current licensing basis at operating plants use the TID-14844 source term and are considered safe to operate. The TID source term should continue to be an acceptable source term for evaluating component integrated dose issues.

COMMENTS ON DRAFT DG-1081 AND DRAFT SRP

CMT #	Page/ Para. #	Comments on DG-1081
1.	Page 4, 1.1.2	Add the following sentence at the end of the first paragraph of 1.1.2, Defense in Depth: "However, reliance on manual operator actions is acceptable if adequate time and information are available to the operator and if the facility modification results in significant hardware simplification (e.g., elimination of a complex automatic control system)."
2.	Page 4, 1.1.1	The alternative source term guidance document should be consistent with the revised 10 CFR 50.59 regulation and associated guidance. This paragraph should reference 10 CFR 50.59 as the source for guidance on margin rather than creating new definitions.
3.	Page 4, 1.1.3, first paragraph	A lack of clarity exists regarding the difference between full implementation and partial implementation of an AST. Paragraph 1.1.3, last sentence, implies that future analyses are to be performed using the AST. This is not true in the case of partial implementation. The following rewording is suggested: "The accident source term is a fundamental input in the design and analysis of plant structures and engineered safety features. Many aspects of facility operation assumed the TID 14844 accident source term. Although a complete re-assessment of the radiological analyses is desirable, the NRC staff is authorizing technically justifiable partial, or selective, uses of an AST if a clear, logical design basis is maintained. For full AST implementation, only the analyses affected by the proposed plant changes require reanalysis (including, as a minimum, the large break LOCA). Non-affected analyses may be left unchanged if found to be adequate. This may create two tiers of analyses, those based on the previous source term (and found to be adequate) and those based on an AST. As a result, the radiological analysis acceptance criteria might be different, with some based on whole body and thyroid dose criteria and some based on the TEDE criteria. The plant design bases should clearly identify where each source term assumption and radiological criteria applies. For partial AST implementation, only the analyses affected by the proposed plant changes need to be reanalyzed. Other, non-affected analyses may be left unchanged. This will create two tiers of analyses, those based on the previous source term (and found to be adequate) and those based on an AST. As a result, the radiological analysis acceptance criteria will be different, with some based on whole body and thyroid dose criteria and some based on the TEDE criteria. The plant design bases should clearly identify where each source term assumption and radiological criteria applies. This paragraph describes two tiers of analyses; those based on the previous source term and those based on the new source term. The paragraph indicates that analyses based on the old source term should be bounding. The paragraph lacks guidance about how the bounding analysis is to be performed. It is unclear why the previous analysis must be bounding relative to the new analysis. The new analysis must only demonstrate it is adequate as compared to the radiological criteria. Revise "bounding" to "adequate."
4.	Page 6, 1.2.1	The second and third sentences are confusing. Revise them to read: "A full implementation revises the plant licensing basis to the AST in place of the previous accident source term model and establishes the TEDE dose as the new acceptance criteria. This applies not only to the analyses performed in the application (which may only include a subset of the plant analyses), but also to all future design basis analyses. At a minimum, when using full implementation of an AST, the DBA LOCA must be re-analyzed using the guidance in Appendix A."
5.	Page 9, 1.3.5	Clarify the regulatory guide to state that currently operating plants choosing to adopt an alternative source term do not need to address EQ doses relating to an increased Cesium releases. This is the subject of a Generic Safety Issue (GSI). Licensees adopting the AST will address the EQ issue in conjunction with all other operating plants based on the GSI outcome. Include an additional sentence at the end of this section as follows:

CMT #	Page/ Para. #	Comments on DG-1081
		"New EQ dose estimates developed to support the modification should use the AST. Note that no special efforts are required to address the impact of the difference in source term characteristics (i.e., AST versus TID) on EQ doses. The NRC, as part of its GSI program, is assessing the effect of increased cesium releases on equipment qualification to determine if licensee action is warranted."
6.	Page 6, 1.2.2	This section states that the NRC staff will "allow licensees the maximum flexibility in technical justified selective implementations...." It provides two examples (release timing delay and eliminating credit for charcoal filtration for the fuel handling accident) to illustrate the point. Revise the text to include other types of selective implementation examples consistent with the desire for maximum flexibility. Two additional examples that should be added are: ● Use of the chemical form of iodine (i.e., I^- vs. I_2) along with pH control in evaluating engineered safety feature (ESF) leakage and iodine release from steam generator tube leakage in main steam line breaks; and ● Modification of the design of a specific component (e.g., control room filter) in a given design basis accident using AST, which does not require reanalysis of other accidents in which this same component is credited.. This is applicable when margin to the dose limits exists in the other design basis accidents and a simple change to the existing radiological analysis is used to reflect the modified component design.
7.	Page 11, 3.1 and 3.2	**Section 3.1** The draft guide states that, "For non-LOCA events, the appropriate radial peaking factor from the facility's core operating limits report (COLR) should be applied." For many events, this would be an appropriate approach since, in general, the fuel rods that would be damaged in a postulated accident involving a reactor transient would be those at a high power level. However, it is well known that the relative power, and thus the radial power peak in a fuel rod, decreases as burnup increases. This is an expected phenomenon since power production must necessarily decrease as fissionable material is consumed. The assumption of a radial peaking factor based on the COLR report therefore should not be a requirement for all analyses since it may be an inappropriate assumption. For example, in the fuel handling accident, the damaged fuel may have been operating at a low power level, far below the peaking factor identified in the COLR – this is particularly true for high burnup fuel. High burnup fuel is expected to have higher fission product gap fractions than lower burnup fuel (see the comment that follows). Consequently, the application of the high radial peaking factor to high burnup fuel results in an unreasonable level of conservatism. It is recommended that the above quoted sentence from the draft regulatory guide be replaced with: For events in which only a fraction of the core is damaged, an appropriately conservative radial peaking factor should be applied to the damaged fuel. For fuel with low burnup (i.e., ≤30,000 MWD/Mtu), the radial peaking factor from the facility's core operating limits report (COLR) should be applied. For fuel with a moderate or high level of burnup, the radial peaking factor may be reduced from the COLR value based on the bounding power history curve associated with the fuel design being used. **Section 3.2** The specification that the gap fractions in Table 3 should be used for all non-LOCA accidents is excessively conservative. The validity of footnote 10 ["The fractions shown in Table 3 are consistent with available data for extended burnup fuel (based on the limiting assembly")], is not evident. The data obtained from fuel rods removed from power reactors support lower gap fractions than those in Table 3. Additionally, the use of a limiting assembly for the determination of the gap fractions results in the inherent assumption that any fuel damaged in a postulated accident is high burnup fuel. This is appropriate only if the level of burnup in the damaged fuel is unknown. This approach does not allow for application of information on the fuel burnup associated with the fuel that would be damaged in a postulated accident. Consequently, it is requested that the paragraph preceding Table 3 be removed:: 'For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. These fractions are applied to the equilibrium core inventory described in Regulatory

CMT #	Page/ Para. #	Comments on DG-1081
		Position 3.1.' In its place, the following is suggested: "For events other than the LOCA with core melt, the fractions of the core inventory assumed to be in the gap for the various radionuclides are dependent on the level of fuel burnup in the damaged fuel rods. The gap fractions for noble gases, iodines, and alkali metals should be as given in Table 3. This table addresses the current upper level for licensed operation of 62,000 MWD/Mtu for the lead rod burnup and the potential for future increases in burnup that may be permitted as fuel designs change. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1." Table 3 Fraction of Fuel Fission Product Inventory in Gap Burnup (MWD/Mtu) — Fraction 0 - 50,000 — 0.0300 55,000 — 0.0425 60,000 — 0.0550 62,000 — 0.0600 70,000 — 0.0800 75,000 — 0.0925 These gap fractions are applicable for fuel damage in accidents for which there is no significant fuel heatup transient (e.g., fuel handling accident, steam line break, steam generator tube rupture, locked rotor). If a transient has a significant fuel heatup rate (e.g., small break LOCA), then an additional two percent of the activity in the damaged rods should be assumed to be released—this is the same as specified in NUREG-1465 for the gap release phase of a large break LOCA that proceeds to core melt. If an applicant chooses not to determine the burnup associated with the fuel damaged in a postulated accident, the analysis should assume that all of the damaged fuel is at the maximum licensed core burnup as is appropriate within the limits of the core design (e.g., if 50% of the core is projected to be damaged and there is no more than 30% of the core that would be above 50,000 MWD/Mtu burnup, then the remaining 20% of the core that is damaged could use the 3% gap fraction). An exception is made for reactivity insertion accidents (rod ejection for the PWR and rod drop for the BWR) because of uncertainties associated with these events and how high burnup fuel will respond during the transient. For the reactivity insertion accidents, the gap fractions for any rods having burnup in excess of 40,000 MWD/Mtu (the NRC's current definition of high burnup fuel) should use the gap fractions in Table 4. The gap fraction of 3% can be used for fuel rods having burnups ≤40,000 MWD/Mtu (consistent with Table 3). Table 4 High Burnup Fuel in a Reactivity Insertion Accident Fraction of Fuel Fission Product Inventory in Gap Nuclide — Fraction I-131 — 0.12 Kr-85 — 0.15 Other Noble Gases — 0.10 Other Iodines — 0.10 Alkali Metals — 0.10 It is noted that the gap fractions here identified in Table 4 are those from the current Table 3 of DG-1081. The above suggestion to use these values does not mean that these are necessarily appropriate. It is industry's understanding that these gap fractions are still being reviewed and that they may be decreasing. With the addition of the new Table 4, the subsequent tables would require renumbering and appropriate corrections made elsewhere for proper referencing of the tables. A more complete discussion of the arguments supporting the above change to DG-1081 is provided in Appendix A.

CMT #	Page/ Para. #	Comments on DG-1081
8.	*Page 12 & 13, 3.2 and 3.3*	These sections identify the LOCA with core melt source term for a large break and the non-LOCA accidents source term. It does not address the small break LOCA for which gap activity releases are assumed. Based on Section 3.2 (non-LOCA accidents), it is assumed that the Table 3 gap fractions are intended to be used. However, the small break LOCA analysis some plants assumes that all fuel rods fail. Table 3 gap fractions assume that the damaged fuel rods are high burnup rods. This is an inappropriate assumption for the situation where all fuel rods are assumed to be damaged. For the small break LOCA, with all rods assumed to fail, the gap fraction should be the same as for the gap release fraction identified in Tables 1 and 2. Failure of a fraction of the fuel rods in the core could not result in a greater release than would failure of all the fuel rods in the core; the source term for the small break LOCA would be no greater than 5% of the core. The onset of the release of gap activity for the small break LOCA will not be the same as for the large break LOCA, but would be determined analytically based on time to uncover the core. After Table 2 add: "The above applies to the large break LOCA with core melt. For the small break LOCA, if all fuel rods are assumed to be damaged, the gap fraction from Table 1 or 2 may be used. If a fraction of the fuel rods is demonstrated to be damaged, the gap fractions from Table 3 may be used. When using the Table 3 gap fractions, the calculated total source term should not exceed the source term associated with failure of all fuel rods in the core." Also add the following after the second sentence in paragraph one of Section 3.3: "For small break LOCAs in which fuel damage is projected, the onset of fuel damage should be consistent with Table 4, unless justified by analysis. The duration of the gap release should be consistent with Table 4."
9.	Page 12, Tables 1 and 2	In NUREG-1465, the gap fraction is identified as being 3% that is immediately released and an additional 2% that is released over the half-hour gap release duration. Modeling the full 5% as being released over the half-hour gap release phase is considered an appropriate alternative since not all of the rods would fail at once. However, as DG-1081 presently is written, the understanding of gap fraction provided by NUREG-1465 is obscured. Industry recommends that a footnote be added referring to the NUREG-1465 material. A suggested footnote would be: "In NUREG-1465, the gap fraction is identified as being 3% initially available in the gap plus another 2% released due to continued heating of the fuel prior to reaching the in-vessel release phase. These are combined in this regulatory guide as 5% to be released over the duration of the gap release phase."
10.	Page 12, Tables 1 and 2	These tables can be misleading unless one is familiar with NUREG-1465. Industry suggests that a third column be added to the tables listing the total release (combination of the gap release phase and the early in-vessel release phase). Alternatively, a note could be added that the Early In-vessel Phase column does not represent a cumulative value, but just the fraction released during that phase.
11.	Page 13, 3.3	The onset of gap release phase for PWRs is given as 10 - 30 seconds without providing a basis for which end of the spectrum is appropriate. From NUREG-1465, the Combustion Engineering plants are reported as being 13 seconds and the Westinghouse plants are reported as being 23 seconds. The onset of gap release for BWRs has been generically approved to be 121 seconds; Table 4 should note these finding. Revise the regulatory guide to reflect the above.
12.	Page 14, 4.	Specific values for dose assessment calculations are provided in Section 4 and the appendices of the draft regulatory guide. For some of these values, no reference or technical basis is provided. Appropriate reference or technical basis should be identified in the final regulatory guide.
13.	Page 14, 4.	Revise this section to clarify the acceptance criteria for use in selective applications. For example, a timing-only application uses a combination of the NUREG-1465 timing with the TID-14844 release fractions and only addresses

<table>
<tr><th>CMT #</th><th>Page/ Para. #</th><th>Comments on DG-1081</th></tr>
<tr><td></td><td>1st paragraph</td><td>noble gas and iodine. The use of only noble gas and iodine implies that the whole body and thyroid limits apply and the change in the timing assumption does not fall under the requirements of 10CFR 50.67. The guidance should clearly recognize that criteria other than TEDE is acceptable in certain circumstances and not subject to the requirements of §50.67.</td></tr>
<tr><td>14.</td><td>Page 15, 4.1.5</td><td>The last sentence is unclear. Replace it with the following:

"The time increments should appropriately reflect the progression of the accident to capture the peak dose interval."</td></tr>
<tr><td>15.</td><td>Page 17, 4.4</td><td>DG-1081 does not consistently define what constitutes a reasonable accident duration for several of the design basis accidents. As an example, the time duration for the LOCA is not addressed in DG-1081. Currently, the 30-day LPZ dose calculated for the Loss-of-Coolant Accident (LOCA) is based on TID-14844 and is not an appropriate reference for the use of an alternative source term. The definition of "duration of accident" for control room and site boundary dose analyses for the various design basis accidents should be included in this regulatory guide.

Revise Table 6 as follows to add a column on "Dose Duration":

<table>
<tr><th>Accident -</th><th>Dose Criteria -</th><th>Dose Duration</th></tr>
<tr><td>LOCA</td><td>25 Rem TEDE</td><td>30 days unless demonstrated shorter by plant design</td></tr>
<tr><td>BWR MSLB</td><td>25/2.5 Rem TEDE</td><td>2 hrs unless demonstrated shorter by plant design</td></tr>
<tr><td>BWR Rod drop</td><td>6.25 Rem TEDE</td><td>24 hrs unless demonstrated shorter by plant design</td></tr>
<tr><td>PWR SGTR</td><td>25/2.5 Rem TEDE</td><td>Until shutdown cooling can remove all decay heat</td></tr>
<tr><td>PWR MSLB</td><td>25/2.5 Rem TEDE</td><td>Unaffected SGs : Until shutdown cooling heat removal rate exceeds decay heat Generation</td></tr>
<tr><td>Affected SGs :</td><td></td><td>Until primary coolant temperature reaches 212</td></tr>
<tr><td>PWR LR</td><td>2.5 Rem TEDE</td><td>Until shutdown cooling can remove all decay heat</td></tr>
<tr><td>PWR REA</td><td>6.25 Rem TEDE</td><td>Containment Scenario: 30 days
Secondary Side release: Until shutdown cooling can remove all decay heat</td></tr>
<tr><td>FHA</td><td>6.25 Rem TEDE</td><td>2 hrs unless demonstrated shorter by plant design</td></tr>
</table></td></tr>
<tr><td>16.</td><td>Page 18, 5.1.2</td><td>DG -1081does not consistently address the issue of Loss of Offsite Power. Appendix F (SGTR) and G (Locked Rotor) list a requirement to assume a coincident LOOP, whereas the other appendices are silent on the issue. In addition, DG-1081 does not acknowledge that most plants have existing licensing basis for their "Loss-of-Offsite Power" assumptions.

Revise the last sentence of Section 5.1.2 to state:

"Assumptions regarding the occurrence and timing of a loss of offsite power should be consistent with the existing licensing basis."</td></tr>
<tr><td>17.</td><td>Page 18, 5.1.3 First Sentence</td><td>Revise the sentence to remove the wording "maximizing the postulated dose." The objective of selecting inputs is not to maximize the postulated dose. When two or more appropriate inputs are available, then the inputs are chosen to ensure that the end result is conservative. Excessive conservatism is not consistent with the NRC performance goal of ensuring its regulatory practices and activities are effective, efficient and more realistic.

Change " maximizing the postulated dose " to "suitably conservative dose."</td></tr>
<tr><td>18.</td><td>Page 19, 5.1.4</td><td>To avoid misunderstanding, the following additional statement should be added to Section 5.1.4 after the sentence "However, prior design bases are unrelated to the use of the AST, or are unaffected by the AST, may continue as the facility design basis."

"Licensees may continue to use site specific models and assumptions unaffected by the AST and previously accepted by the NRC staff, even though they may be different from those listed in DG-1081 and its Appendices. This includes but is not be limited to assumptions with respect to single failure, passive failure, and amount of ESF leakage, iodine spiking, etc."</td></tr>
<tr><td>19.</td><td>Page 20, 6.</td><td>"Duration of accident" for radiological equipment qualification analyses needs to be included in Section 6.

The regulatory guide does not provide guidance as to what constitutes reasonable accident duration. Currently this is</td></tr>
</table>

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		left to the licensee and has resulted in durations that vary from two months to one year. As an example, though, there is no technical basis for the recirculation spray pumps to be available at one plant for a year whereas at another plant it is required for 60 days. In addition, no credit is given for additional backup equipment that can be brought on-site and utilized for maintenance of safe shutdown after a reasonable amount of time has passed since accident initiation. The guidance should distinguish between the "mitigation phase" and the "recovery phase" of the accident. The mitigation phase should be defined as the "accident duration" used in accident analyses (i.e., for site boundary & control room). The "recovery phase" should be defined as the period following the mitigation phase during which additional cleanup and recovery equipment can be brought on site and credited for maintenance of safe shutdown. This latter equipment need not be qualified as safety-related components. The industry is proposing that the "duration of accident" used for *radiological equipment qualification* purposes be similar to that used for control room and site boundary dose calculations (i.e., for the LOCA it should be 30 days). This position is reasonable for the following reasons: • TMI experience demonstrated that an entire safety-related RHR system including associated structures for housing the referenced equipment could be installed by a licensee within seven days of the event. • The NRC has previously evaluated accident radiation dose for equipment qualification purposes. NUREG/CR-5313 (SAND-88-3330), "Equipment Qualification Risk Scoping Study," concluded that: (1) the importance of the accident radiation dose is overemphasized, (2) that equipment qualification issues associated with long term accident equipment operability are not risk significant, and (3) equipment qualification should focus on ensuring equipment operability for the first few days of the accident exposure, as illustrated by plant risk assessments. • The crossover point between equipment qualification doses predicted by the AST versus TID for equipment exposed to post-LOCA recirculating fluids is approximately 30 days. A specified 30-day integration period for equipment qualification purposes supports the technical adequacy of either source term as a licensing basis. It also diminishes the potential for a future safety concern relative to the qualification of safety-related equipment for plants that retain the TID source terms as their design basis. In summary, Section 6 of DG-1081, (which states that it replaces Regulatory Guide 1.89 in providing guidance for dose assessments associated with equipment qualification), should be updated to provide guidance on "duration of accident" for equipment qualification dose analyses. In addition, DG 1081 should state that the "duration of accident" utilized for *radiological equipment qualification* purposes should be similar to that utilized for control room and site boundary dose calculations (i.e., for the LOCA it should be 30 days), and that licensees using AST may revise their licensing basis "duration of accident" to reflect this guidance. Revise Section 6 of DG-1081 to include the following: "From an equipment qualification perspective, the time after an accident may be considered to be divided into the mitigation phase and the recovery phase. The mitigation phase (defined as the first 30 days for a LOCA) is the time immediately after the event when existing plant design/response has to be dependent on to mitigate the event; whereas the recovery phase is the period after the mitigation phase during which additional cleanup/recovery equipment can be brought on site, as needed, and credited for maintenance of safe shutdown. Safety related equipment must be qualified to withstand the environment to which it is exposed to during the mitigation phase. Licensees adopting ASTs may revise their licensing basis "duration of accident" to reflect this guidance. "
20.	Page A-1, 2.	The regulatory guide should address when iodine re-evolution occurs. Add the following sentence after " ... fission products should be assumed to be in particulate form": "Iodine re-evolution in elemental form may need to be addressed if the pH of the recirculation fluid is less than seven."
21.	Page A-2, Footnote 1	This footnote states that the elemental iodine decontamination factor (DF) should be based on the amount of elemental iodine that is airborne at the end of the early in-vessel release phase rather than the total release of elemental iodine. This approach is nonconservative. The partitioning between the sump and the atmosphere would be based on the total amount of elemental iodine released. Revise the footnote to reflect this.

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22.	Page A-4, 5.1	The amount of activity entering the sump water is not clearly defined. Revise the first sentence to read: "With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core."
23.	Page A-2, 3.3	Please clarify the restrictions on allowable DF for aerosol removal by sprays. Add the following paragraph to the two existing paragraphs in Section 3.3 "Note that when using SRP 6.5.2 methodology, the particulate removal rate must be reduced by a factor of 10 when a DF of 50 is reached. This reduction of the removal rates is not required when the release coefficients are based on the calculated time dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays."
24.	Page A-3, 3.8	Section 3.8 of Appendix A states that "For BWRs with Mark III containments, the flow rate from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation." Such flows are also applicable to other BWR containment designs; i.e., there is nothing unique about Mark III containments with respect to such flows since the flows originate in-vessel. The words "with Mark III containments" should be deleted and the words "wetwell or torus" should be added after "primary containment." DG-1081 should simply state that flows will exist between the drywell and the wetwell/containment during core degradation, and that appropriate credit will be given for such flows in the application of the AST to BWR plants. Also, if analyses justify that an uncovered core could not sustain a two-hour release of the magnitude given in NUREG-1465 without some degree of core debris relocation (or of limited coolant injection and associated steaming), then the limitation placed on consideration of such core debris relocation (or additional steaming) should be lifted.
25.	Page A-3, 3.9	Expand the comment on purging to acknowledge the variation in purge practices, e.g., continuously, once per month, prior to any planned containment entry, as needed, etc. to reduce containment pressurization.
26.	Page A-6, 6.3	Revise the last sentence of Appendix A, Section 6.3 to clarify that it is not excluding slug flow. State: "Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used, if justified."
27.	A-6, 7.	Appendix A states if purging is part of the design basis, then dose consequences for post-LOCA primary containment purging as a combustible gas control measure should be added to the other release paths. However, some plants have purging requirements after 30 days (i.e., after the duration of the accident). Consequently, the following clarification should be included in the guidance. Revise the last sentence in Section 7 to read: "If primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA."
28.	Page B-1, 1.3	Paragraph 1.3 of Appendix B states that the iodine release from the fuel in a FHA should be 99.75% elemental and 0.25% organic. In fact, the iodine released from the gap of spent fuel will be almost entirely CsI or some type of I-. The basis for this is given below. It is recognized that spent fuel pool pH will need to be considered just as containment sump pH must be. However, the form of iodine released from the fuel should be recognized and correctly stated in Appendix B. The chemistry of fission products in a fuel rod under normal reactor operation was reviewed to document what is known about the chemical form of iodine in the fuel-cladding gap. A comprehensive discussion of this subject appears in Section 4.1, titled "Fission Product Behavior in Fuel," of NUREG-0772 [1]. In this discussion there is mention of one direct observation relating to the chemical form of iodine in the fuel cladding gap: crystalline deposits containing cesium and iodine on internal cladding surfaces reported by Cubicciotti and Sanecki [2]. Indirect observations of CsI in the gap are provided by two ORNL experiments in which measurements of fission product release and transport from irradiated fuel rod segments were made at low temperatures (<1500 K) in flowing

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		helium [3]. A fuel rod irradiated in H. B. Robinson was used in one experiment and in the other experiment, a rod irradiated in Peach Bottom-2 was used. In each experiment, the fraction of the inventory of iodine released was essentially the same as that of cesium and very nearly equal to the fraction of fission gas inventory measured (by rod puncturing, collection and analysis) to have been released during irradiation. Quite different values of krypton release during irradiation were measured in the two rods (0.3% in the H. B. Robinson rod and 13.5% in the Peach Bottom-2 rod) due to large differences in linear heat generation rate (170 W/cm in H. B. Robinson and 300 W/cm in Peach Bottom-2). In both experiments, the vast majority of the iodine and cesium released deposited in a thermal gradient tube at a region in the temperature range 623 - 773 K. Similar condensation profiles were obtained in control tests with CsI. The excess cesium (~10:1 mass ratio Cs/I) in both tests behaved as the cesium in a test in dry air where the likely chemical form was identified as an oxide, less volatile than elemental cesium, probably Cs_2O. In the test with H. B. Robinson fuel 91.2% of the iodine behaved like CsI and only 0.27% as I_2 (captured in impregnated charcoal). In the test with Peach Bottom-2 fuel, 99.99% of the iodine behaved like CsI and only 0.004% as I_2. Evidence from the ORNL gap release experiments [3] clearly eliminates the possibility of the elemental forms of iodine and cesium in the fuel-cladding gap. The boiling point of iodine is 456 K and that of cesium is 963 K. The heat up rates in the two tests were relatively slow (28 K/min for the H. B. Robinson fuel and 11 K/min for the Peach Bottom-2 fuel), however the release of cesium was detected only when fuel rod segment temperatures reached 873 K (90% of the boiling point). In addition, there was ample time for the release and transport of elemental iodine through the thermal gradient tube to the charcoal trap without interaction with other reactive fission products (e.g., cesium with its much higher boiling temperature), but well less than 1% of the iodine released was found in the charcoal. Campbell, Malinauskas and Stratton, in an earlier paper [4], also concluded that the ORNL gap release experiments eliminated elemental iodine as the chemical form of iodine in the gap based on the deposition of iodine at temperatures well in excess of the boiling point of iodine. They also point out that studies of iodine redistribution in test fuel rods indicate a tendency of fission product iodine to migrate within the fueled region, but not beyond the top of the fuel column into the gas plenum region, behavior indicative of a chemical form less volatile than elemental iodine. The above experimental evidence against elemental iodine and in support of CsI in the gap is in agreement with the results of a thermodynamic study by Besmann and Lindemer [5] which indicates that CsI is the preferred chemical form of iodine in the gap of a fuel rod under reactor operating conditions. One must bear in mind that kinetic effects may not permit thermodynamic equilibrium to be reached and thermodynamic results depend on the thermodynamic data and chemical species input to the analysis. Besmann and Lindemer calculate that the likely form of cesium in the fuel is Cs_2UO_4 over which the cesium partial pressure is significant, leading to the transport of cesium to the gap where it reacts with iodine to form CsI. Iodine does not interact chemically with the fuel and can be assumed to transport to the gap in elemental form, I or I_2. Both iodine and cesium are likely transported through the fuel via fission gas bubbles. Indeed, as shown by the ORNL gap release experiments, the fractions of the inventories of fission gas, iodine and cesium in the gap are essentially identical. The Besmann and Lindemer study did not include the species Cs_2ZrO_3 because thermodynamic data for this species were not available at the time the study was performed. The Gibbs free energy for this species was calculated in Rev. 1 of the VICTORIA code description document [6] to be nearly as stable as Cs_2UO_4. To the extent that cesium forms Cs_2ZrO_3 in the gap, the cesium availability to form CsI is reduced. However, the molar ratio of cesium to iodine in the gap is approximately ten. The ORNL gap release experiments indicate that iodine exists as CsI in the gap no matter what other chemical forms of cesium exist in the gap, such as Cs_2ZrO_3 or Cs_2O. Besmann and Lindemer find that zirconium iodides such as ZrI_3 may form but are much less important than CsI. Elemental or molecular iodine is found to be insignificant in the gap. In conclusion, experimental and thermodynamic evidence exists for the strong likelihood that CsI is the preferred form of iodine in the fuel-cladding gap. Conversely, experimental and thermodynamic evidence eliminates the possibility of any significant presence of elemental or molecular iodine in the gap. <u>References</u>: 1. U. S. Nuclear Regulatory Commission, "Technical Bases for Estimating Fission Product Behavior during LWR Accidents," NUREG-0772, June 1981. 2. D. Cubicciotti and J. E. Sanecki, J. Nucl. Mater., 78, page 96 (1978). 3. J. L. Collins, M. F. Osborne, R. A. Lorenz, and A. P. Malinauskas, Fission Product Iodine and Cesium Release Behavior Under Light Water Reactor Accident Conditions", Nucl. Technol. 81, page 78 (1988).

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		4. D. O. Campbell, A. P. Malinauskas, and W. R. Stratton, "The Chemical Behavior of Fission Product Iodine in Light Water Reactor Accidents", Nucl. Technol. 53, page 111 (1981). 5. T. M. Besmann and T. B. Lindemer, "Chemical Thermodynamics of the System Cs-U-Zr-H-I-O in the Light Water Reactor Fuel-Cladding Gap", Nucl. Technol. 40, page 297 (1978). 6. T. J. Heames, et al., "VICTORIA: A Mechanistic Model of Radionuclide Behavior in the Reactor Coolant System Under Severe Accident Conditions", NUREG/CR-5545, SAND90-0756, Rev. 1 (December 1992).
29.	Page B-2, 4.3	Section 4.3 states that radioactivity release from the fuel pool after a FHA in the fuel building should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. Revise this to indicate that if mixing can be demonstrated, credit for mixing/dilution in the fuel building can be taken following a FHA in the fuel building. This is consistent with section 5.5 that specifically addresses credit being allowed for dilution for a FHA in containment.
30.	Page B-2, Footnote 3:	The footnote specifies limitations on plant operation that may or may not be required of a specific facility. The assumption that administrative controls will be required to isolate the containment in 30 minutes should not be specified. Revise the footnote to read: "If there are administrative controls to close the air lock or hatch in less than two hours, the radiological analyses may credit plant-specific containment isolation practices, if justified."
31.	Page E-1 and F-1, 2.1 and 2.2	Revise paragraphs 2.1 and 2.2 to allow use of alternatives to pre-accident spike of 60 micro-Ci/g and coincident spike of 500 if applicant presents reasonable evidence for other values. Evidence exists which shows that when activity levels are low, the spiking multiplier may be high; but that when activity levels are more representative of Technical Specification limits (i.e., of the same order of magnitude), then the spiking multiplier is observed to be commensurately lower.
32.	Page E-2, F-2, H-2	Appendices E, F, and H of DG-1081 specify that iodine released via the steam generators should be assumed to be 97% elemental and 3% organic. This statement should be revised to state that it applies to the iodine which is released via the steam generator(s) to the environment per Attachment [X], the form of the iodine released from the fuel gap to the coolant (and thus from the steam generator primary side to the secondary side) is primarily CsI.
33.	Page E-2, 5.5	Paragraph 5.5 states that "all iodine and particulate radionuclides released from the primary system via the faulted steam generators should be assumed to be released to the environment with no mitigation. This is overly conservative. There are several mechanisms which will retain iodine and other particulates (including iodine retained in liquid remaining after the flash, I- retained on tube metal surfaces during evaporation to dryness of remaining liquid, and I_2 deposition on tube surface) and these mechanisms should be allowed when justified by the licensee. Revise this statement to insert "unless a detailed mitigation (i.e., removal) model is proposed and accepted" after "mitigation."
34.	Page E-1, 2.2 (Also applicable to Page F-1, 2.2)	The iodine spike duration is stated to have an 8-hour duration. From calculations that have been performed, an 8 hour long spike can result in the release of more activity to the primary coolant than would be available in the fuel-clad gap for the leaking fuel rods. Revise the text to read: "The assumed iodine spike duration should be 8 hours unless a shorter duration can be technically justified."
35.	Page E-3, 5.8 (also Page F3, 5.8; page G-2, 5.8; and page H-2, 7.6)	The issue of steam generator tube uncovery for short periods is raised with the statement that, "Primary-to-secondary leakage that occurs during these periods should be assumed to be released to the environment without mixing in the steam generator bulk water and no credit should be taken for iodine partitioning." This guidance should more concisely focus on the flashed fraction of the primary-to-secondary leakage. Additionally, the proposed guidance is contrary to the NRC position taken with respect to Westinghouse steam generators (NRC letter of 3/10/1993 from Robert C. Jones, Chief, Reactor Systems Branch to Lawrence A. Walsh, Chairman, Westinghouse Owners Group). It is suggested the following sentences replace the second sentence in the draft paragraph: "Based on the conservation of energy principle, only a portion of the fluid that leaks from the primary coolant system to the secondary side of the steam generator flashes to steam. The portion of the leakage that remains as liquid returns to the bulk water due to gravity and/or impaction on steam generator internal components (e.g., other tubes, dyers, separators). The flashed portion of the primary-to-secondary leakage should be assumed to be

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		released to the environment without mixing in the steam generator bulk water and with no credit taken for iodine partitioning. This guidance does not apply to steam generator designs for which it has been demonstrated that the impact of tube uncovery is negligible."
36.	Page E-2, Footnote 3	Delete this footnote. Primary-to-secondary leak rate is described in item 5.1.
37.	Pages G-1, 2	It is inappropriate to reference the steam generator tube rupture (SGTR) in this appendix. A locked rotor accident with no fuel damage is not similar to the SGTR. Comparison with the steam line break outside containment is more appropriate.
38.	Page H-1, 4	**Containment Sprays during REA**: The following changes are suggested: • Replace the word *LOCA* by *REA*. • Add a statement that evaluation of sump pH following a REA is unnecessary if no credit is taken for iodine removal from the containment atmosphere.
39.	Page H-2, 7.3	The existing text should be more precise. The draft language could be interpreted to mean that all noble gas activity entering the containment atmosphere is released to the atmosphere. Revise "All noble gas radionuclides released from the primary system" to "All noble gas radionuclides released to the secondary system."
40.	Page I-2, 3	The guidance should allow an appropriate fraction of the activity initially released to the drywell to be transported from the drywell to containment since this will reduce the release to the secondary building. Revise the text to permit this.
41.	Page I-2, 8	This section states that gamma dose rates should be multiplied by a correction factor of 1.3 to account for the omission of the contribution from the decay chains of the isotopes. Please clarify this guidance to note that the correction factor should only be applied if the licensee's methodology/computer code does not explicitly account for decay chain doses.

Appendix A

FISSION PRODUCT CONTENT IN THE FUEL ROD GAP

Introduction

The NRC alternate source term (AST) report (NUREG-1465) [1] states that for LOCAs an appropriate value for noble gas and halogen fission product content in the fuel rod gap would be 5% (3% initial release and an additional 2% due to heatup), based on a review of previous research and analysis. Furthermore, NUREG-1465 reported that a value of 3% could be used for events for which fuel cooling was maintained (e.g., the fuel handling accident (FHA) or a LOCA in which core cooling is maintained). The System 80+ design certification program used a value of 5% for LOCA and all non-LOCAs. The AP600 design certification program used a value of 5% for the LOCA, but assumed a value of only 3.6% for the non-LOCA events. The value of 3.6% was derived by multiplying the 3% value by a factor of 1.2 to account for high-burnup effects.

In the NRC effort to allow the use of the AST for design basis accident (DBA) analysis of operating reactors, NRC staff proposes in Table 3 of draft Regulatory Guide DG-1081 that the NUREG-1465 gap fractions be used for LOCA, but that the following, more conservative, assumptions for fission products in the fuel rod gap be used for non-LOCA events:

I-131	12%
Kr-85	15%
Other iodines	10%
Other noble gases	10%
Alkali metals	10%

It is industry's understanding based on a review of NUREG-1465 and on recent discussions with NRC staff that this increase in gap fractions has been proposed for non-LOCA events for two reasons: (1) concern about recent test data on gap release in reactivity insertion accidents, and (2) concern about increased gap release for fuel irradiated beyond 40,000 MWd/MTU.

While acknowledging the NRC concerns (see further discussion below), industry believes that the formulation in NUREG-1465 for non-LOCA DBAs is still generally applicable, and that the values proposed in DG-1081 are excessively conservative (except possibly for reactivity insertion accidents). The NRC concerns and associated industry proposed alternatives to DG-1081 are addressed below.

Reactivity Insertion Accidents

Industry recognizes that design basis Reactivity Insertion Accidents (RIAs) (i.e., PWR rod ejection or BWR rod drop) present the potential for power excursions and associated rapid change in local fuel and cladding conditions (e.g., fuel temperature, cladding stress and strain, fuel rod pressure). The unique nature of these transients and their localized behavior may warrant the assumption of a fission gas release fraction that is greater than that assumed in the radiological consequence analysis of other non-LOCA events. Experimental simulations at both the French CABRI and Japanese NSRR facilities, have produced results that indicate a potential for significant fission gas releases from high-burnup fuel during RIA events. This research is continuing and involves the participation of international organizations, as well as NRC and EPRI, through its Robust Fuel Program. Until sufficient information becomes available to resolve this issue, industry recommends retaining the values currently proposed in DG-1081 (see the above table) for high-burnup fuel damaged in RIA events. Industry expects to reassess the proposed high-burnup fuel gap fraction values for RIA events when the results and interpretation of the ongoing research becomes more conclusive.

Industry recommends that DG-1081 be revised to indicate that the Table 3 gap fractions should be applied to high-burnup (>40,000 MWD/MTU) fuel, but that fuel defined as not having high burnup may use the gap fractions as defined in the following sections. Further, DG-1081 could state that if the burnup of damaged fuel rods is unknown, all damaged fuel rods should be considered as high-burnup fuel.

Gap Fraction vs. Burnup

One of the issues which bears on gap release for high-burnup fuel is how gap fraction changes with increasing burnup. The discussion in this section applies directly to accidents in which long-term cooling is maintained (e.g., the fuel handling accident, steam generator tube rupture, or steam line break). For in-core, non-LOCA events in which long-term cooling is not maintained, the gap fractions defined in this section may need to be increased to reflect the additional releases associated with fuel heating (depending on the extent of fuel pellet heating). The increase in gap activity releases due to fuel heating is discussed below in the section entitled, "Increase in Gap Fraction from Post-Accident Heating."

Industry recognizes that gap fraction can increase with increasing burnup and believes that a reasonably conservative estimate of the fission products in the fuel rod gap can be made by bounding the measurements of the percent fission gas release in fuel rods taken from operating reactors. Recent measurements of volatile fission product content in the fuel rod gap for high-burnup fuel have been published by EPRI [2], and similar data have been presented in other reports [3, 4]. The data cover a range of fuel rod designs and fuel designers, and burnups range from about 20,000 MWD/MTU to about 64,000 MWD/MTU.

The combined data of references [2-4] show that fission product release is less than 1.0% up to a burnup of about 30,000 MWd/MTU. As the burnup increases, the gap fission product content increases to about 2% at 40,000 MWd/MTU. At about 50,000 MWD/MTU, the gap fraction increases with burnup at a rate of about 2.3% per 10,000 MWd/MTU per reference [2]. In the table below, the middle column shows an envelope of the data. The right-hand column of the table shows the industry recommended gap fractions for use in analyzing non-LOCA events. The degree of conservatism of this proposed envelope relative to the data is illustrated in Figure A-1.

Rod Average Burnup (MWD/MTU)	Envelope of Measured Fission Gas Release (%)	Industry Proposed Gap Fraction for Use in Design Basis Analysis (%)
0	0.0	3
20,000	1.0	3
30,000	1.0	3
40,000	2.0	3
50,000	2.0	3
60,000	3.5	5.5
62,000	3.8	6.0
70,000	5.0	8.0
75,000	5.8	9.25

The quantity of volatile gas in the free volume region of each fuel rod was obtained by puncturing the rods long after reactor shutdown such that short-lived fission gases decayed before those measurements. As indicated in Table 5-5 of the reference EPRI report (TR-103302-V2), the krypton and xenon isotopes captured from the punctured fuel rods are principally stable isotopes, although some Krypton-85 is captured (due to its relatively long half-life). As shown in Table 5-6 of the EPRI report, the quantity of xenon plus krypton collected is compared to the quantity of xenon plus krypton generated in the particular rod to determine the percentage of fission gas released from the fuel pellet column for the stated burnup. Each datum on the graph in Figure A-1 reflects the percent of xenon plus krypton gas that migrated from the fuel pellet column to the free volume (i.e., the "gap") of that particular fuel rod. This "migration percentage" includes stable and long-lived isotopes.

A bounding envelope that industry proposes to use for the safety analyses was drawn over the data with margin added for conservatism. Industry believes that the bounding envelope is conservative for the following reasons:

- The envelope is drawn well above all data (approximately 100% margin above the mean values and 50% margin above the maximum values).

- The xenon plus krypton release percentage would be applied to iodine isotopes, which would be conservative since iodine is less volatile than xenon and krypton.

Values taken from the bounding envelope of fission gas release vs. burnup for use in safety analyses are applied to long-lived radioactive isotopes and short-lived isotopes such as Xe-133. The application to short-lived isotopes inherently assumes that short-lived isotopes migrate to the gap at the same rate as stable and long-lived radioactive fission gas isotopes.

The reference data have been extrapolated to 75,000 MWd/MTU in order to encompass the burnup range which industry anticipates could be utilized over the next decade or so. This is a modest extrapolation of the above-referenced data. The 62,000 MWD/Mtu burnup data point is included in the table below since this is currently the maximum licensed burnup for operating plants.

Industry proposes that DG-1081 be changed to specify that for postulated accidents which may have damaged fuel but do not have a fuel heatup, licensees should utilize the gap fractions as a function of burnup as specified in the right-hand column of the above table.

Increase in Gap Fraction from Post-Accident Heating

The following discussion applies to in-core events in which there is significant fuel heatup or long-term cooling is not maintained (excluding the high-burnup fuel rods damaged in a reactivity insertion accident, which are discussed above).

It is recognized that if fuel experiences heatup due to a transient, some additional fission gas may be released from the pellet to the reactor coolant through the failed cladding. This was explicitly addressed in NUREG-1465 for a LOCA. It is also true for non-LOCA events; however, the degree of heatup and corresponding fission gas release is a function of the postulated accident being analyzed. For some accidents there is little or no fuel heatup (e.g., fuel handling accident, locked rotor, steam generator tube rupture, main steam line break) and, hence, there would be no transient fission gas release from the fuel pellets.

The fraction of fission product activity that would be released due to holding fuel at a temperature of 1200 °C (2192 °F) for a period of ten minutes was modeled and reported in reference [5] with the determination that 2.8% of the krypton would be released, less than 1.0% of the xenon would be released, and less than 0.1% of the iodine and cesium would be released.

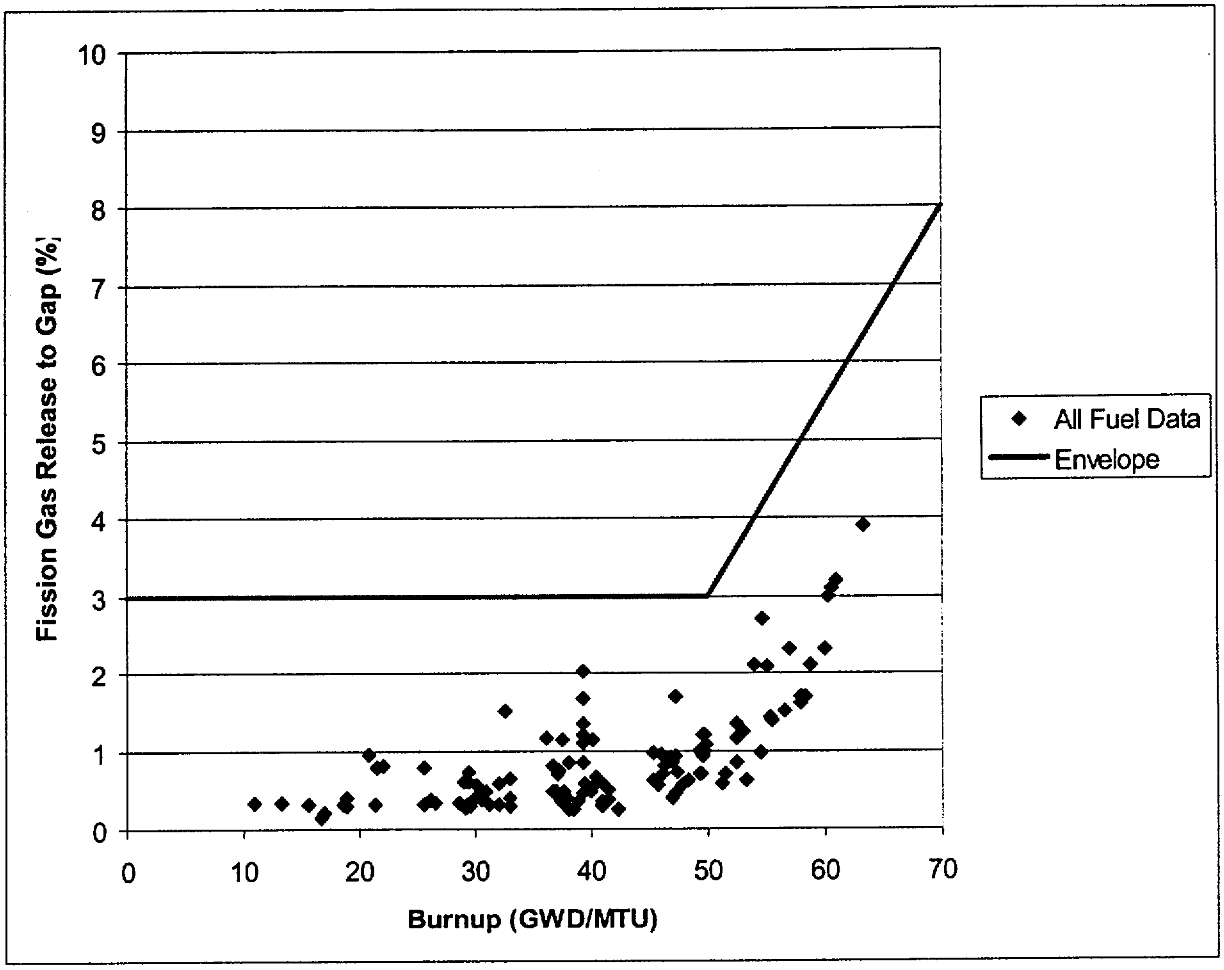
Industry proposes that release of fission products to the fuel clad gap during non-LOCA events be addressed as follows:

- if it is known or demonstrated that there is little or no fuel heatup, no transient fission product release would be assumed, and
- if it is expected that some sustained fuel heatup would occur, then the assumed transient fission product release would be the same as that identified in NUREG-1465 for the design basis LOCA. That is, an additional 2% of the fuel rod fission gas, iodines, and cesiums are assumed to enter the fuel rod gap and be available for release from the damaged rods.

References

1. L. Soffer et al., "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.
2. G. Smith et al., "Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1," EPRI report TR-103302-V2, July 1994.
3. "Extension of the 1-Pin Burnup Limit to 65 MWd/kgU for ABB PWR Fuel with OPTIN™ Cladding," ABB report CENPD-388-P, Figure 2.2.2.2-1.
4. S. R. Pati et al., "Fission Gas Release from PWR Fuel Rods at Extended Burnups," Proceedings of the American Nuclear Society Topical Meeting on Light Water Reactor Fuel Performance, Orlando, Vol. 2, Pages 4–19, April 21–24, 1985.
5. H. P. Nourbakhsh et al., "Fission Product Release Characteristics into Containment Under Design Basis and Severe Accident Conditions," NUREG/CR-4881, March 1988.

Figure A-1: Measurements of Fission Gas Content in LWR Fuel Rods



CMT #	Page/ Para. #	Editorial Comments on DG-1081
1.	Page 2, first paragraph of Part B	The word "assess-ments" should be "assessments".
2.	Page 7, 1.3.2 first paragraph	Midway through the paragraph there is a sentence that is terminated with two periods. The number of evalautions differd\rs from the first to the secon sentence, revise the firstn sentence to a singilar evalaution.
3.	Page 13, Table 5	The last entry is incorrectly formatted. The columns don't line up.
4.	Page 17, 4.2.7	The equation should have a reference to the Murphy-Campe report (Reference 20).
5.	Page 19, 5.3, Last paragraph	In the last sentence there is "?/Q" which should be "χ/Q".
6.	A-3, 3.8:	In two places "radioactivity" is written as "radioac-tivity".
7.	B-1, 2	"decontamination" is written as "decontamina-tion".
8.	Page E-2, 5.5	The reference should be to a single faulted steam generator, not plural.
9.	Page E-1, 2.1	60 Ci/gm should be 60 micro-Ci/gm (comment also applies to page F-1)
10.	Page E- 3, 5.8	The word "need" should be "needs" (this is the fourth word from the end).
11.	Page H-1, 4	"4.85&" should be "4.85%".
12.	Page I-1, 3	Should the reference to "Appendices B through G" be instead to "Appendices B through H"?

Comments on Draft Standard Review Plan (SRP) Section 15.0.1

<table>
<tr><th>CMT #</th><th>Page/ Para. #</th><th>Comments on Draft SRP</th></tr>
<tr><td>1.</td><td>Pages 15.0.1-4 Item 3 and 15.0.1-12 Item 5</td><td>These two sections state that the core inventory should be based on rated thermal power, enrichment and burnup. The codes used in DBA accident analyses, such as TACT and RADTRAD, determine core inventory solely as a function of rated thermal power. Reference to the other two parameters should be deleted from the SRP.</td></tr>
<tr><td>2.</td><td>Page 15.0.1-6, Table 1</td><td>The definition of "duration of accident" for control room and site boundary dose analyses for the various design basis accidents needs to be included in the SRP. Recommended action: Update Table 1 of Section III to include a column on "Dose Duration":
<table>
<tr><th>Accident -</th><th>Dose Criteria -</th><th>Dose Duration</th></tr>
<tr><td>LOCA</td><td>25 Rem TEDE</td><td>30 days unless demonstrated shorter by plant design</td></tr>
<tr><td>BWR MSLB</td><td>25/2.5 Rem TEDE</td><td>2 hrs unless demonstrated shorter by plant design</td></tr>
<tr><td>BWR Rod drop</td><td>6.25 Rem TEDE</td><td>24 hrs unless demonstrated shorter by plant design</td></tr>
<tr><td>PWR SGTR</td><td>25/2.5 Rem TEDE</td><td>Until shutdown cooling can remove all decay heat</td></tr>
<tr><td>PWR MSLB</td><td>25/2.5 Rem TEDE</td><td>Unaffected SGs: Until shutdown cooling heat removal rate exceeds decay heat generation</td></tr>
<tr><td>Affected SGs :</td><td></td><td>Until primary coolant temperature reaches 212 F</td></tr>
<tr><td>PWR LR</td><td>2.5 Rem TEDE</td><td>Until shutdown cooling can remove all decay heat</td></tr>
<tr><td>PWR REA</td><td>6.25 Rem TEDE</td><td>Containment Scenario : 30 days
Secondary Side release : Until shutdown cooling can remove all decay heat</td></tr>
<tr><td>FHA</td><td>6.25 Rem TEDE</td><td>2 hrs unless demonstrated shorter by plant design</td></tr>
</table>
</td></tr>
<tr><td>3.</td><td>Page 15.0.1-8, III,2.b.(1)</td><td>III.2.b.(1), the first sentence reads:

"A selective implementation on the basis of only the timing characteristic of an AST will normally be found to be acceptable without dose calculations, provided other impacts, if any, are adequately dispositioned."

This statement is correct if "timing only" is interpreted as being the timing of the onset of core activity releases for the LOCA. Another interpretation of "timing only" might be to have the TID-14844 core release be not only initiated at a delayed time but also include the time required to release core melt activity to the containment as identified in Table 4 of draft regulatory guide DG-1081.

Revise the cited sentence to read:

"A selective implementation on the basis of only the timing characteristic of an AST may be acceptable without dose calculations, provided other impacts, if any, are adequately dispositioned."

Alternatively, the following wording would also remove the ambiguity:

"A selective implementation on the basis of only the AST timing of the onset of core releases for the LOCA will normally be found to be acceptable without dose calculations, provided other impacts, if any, are adequately dispositioned."</td></tr>
</table>

CMT #	Page/ Para. #	Comments on Draft SRP
4.	Page 15.0.1-12, III.6.b	Add the following statement after "...departures from this guidance will warrant additional review." "Licensees may continue to use site specific models and assumptions unaffected by the AST and previously accepted by the NRC staff, even though they may be different from those listed in DG-1081 and its Appendices. This includes but is not be limited to assumptions with respect to single failure, passive failure, and amount of ESF leakage, iodine spiking, etc."
5.	Page 15.0.1-13, III.8	The references should be revised to indicate that that currently operating plants choosing to adopt an alternative source term do not need to address equipment qualification insights relating to an increased Cesium releases. This is the subject of a Generic Safety Issue (GSI). Licensees adopting the AST will address the EQ issue in conjunction with all other operating plants based on the GSI outcome. The text in the SRP should be consistent with the equivalent text implemented in the issued regulatory guide.