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1.	Page 4, 1.1.2 DG-1081	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)."	The NRC staff believes that the proposed DG language is appropriate. While the simplifications described in the proposed language could be appropriate in limited circumstances, reliance on manual operator actions as a compensation for design deficiencies is generally unacceptable.
2.	Page 4, 1.1.1 DG-1081	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.	It is important to note that 50.59 does not apply to amendment requests under 50.90 (and 50.67). The staff included this section to remind applicants that safety margins are required to be maintained for modifications that wouldn't be reviewed under 50.59. The language of this section was reviewed by the staff for consistency with the new 50.59 rule and guidance. The guide recognizes that the licensee will use the guidance for 50.59 in assessing impact on margins in AST applications subsequent to the initial approved application. A reference to 50.59 for use with subsequent modifications was added.
3.	Page 4, 1.1.3, first paragraph DG-1081	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 staff agrees that the proposed DG language could be clearer. However, the staff disagrees with the proposed language. Considering a full implementation first: In a full implementation, the licensee proposes to replace the current source term with an alternative source term. Once the AST is approved, the staff expects the licensee to use it in all future analyses. The staff recognizes that there may be some analyses which can be shown to be bounding with the prior source term and dose criteria. The staff will not require these analyses to be revised as part of the initial submittal. However, the source term contained in those analyses is no longer valid, having been replaced by the proposed and approved AST. The licensee's evaluation demonstrated that the results of those analyses would continue to be bounding with the new AST and dose criteria, in effect replacing the prior source term. While the staff does not expect all of these affected analyses to be updated for the initial submittal, it does expect that, if the any of the analyses are updated in the future for whatever reason, the re-analysis (on an analysis-by-analysis basis) will use the AST and the dose criteria which are now the facilities licensing basis. This philosophy applies to a selective implementation, but only to those analyses within the boundaries of the approved selective implementation. The affected text in the final guide has been revised to clarify the staff position.

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		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 DG-1081	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."	The affected text in the final guide has been revised as suggested.
5.	Page 9, 1.3.5 DG-1081	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: "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."	Although the staff agrees with the comment, the staff does not wish to require the use of the AST. The staff has revised the section to allow licensees to use either the AST or TID14844 pending outcome of the GSI. The staff included the caveat regarding "special efforts," but restated it as "plant modification" for clarity. Similar text added to position C.6.
6.	Page 6, 1.2.2 DG-1081	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.	For the first example, the NRC staff does not agree with crediting chemical form without considering the changes in isotopic release fractions. For the second example, the staff does not believe that this approach supports a clear design basis. The staff expects licensees to consider the impact of a modification on all affected analyses and to update those analyses appropriately.
7.	Page 11, 3.1 and 3.2 DG-1081	**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.	The NRC staff has revised this section of the regulatory guide, reflecting many of the changes advocated by the industry. Generally, the NRC staff agrees that the gap fractions in the draft guide were greater than necessary. These were the best data available to the staff at that time. Since the draft guide was published for comment, the staff has completed work on gap fractions for

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		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. **<u>Section 3.2</u>** 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 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	extended burnup fuel. Nonetheless, the staff believes that there is a great deal of uncertainty in the projection of gap fractions, particularly for extended burnup fuel. The staff considered the information provided by the industry in Appendix A to the comments. The staff's concerns are as follows: • While the industry position is based largely on actual sampling of fuel, the majority of the data points are for low burnup fuel and there are no data points above about 62 GWD/MTU for which to establish a slope with reasonable certainty. The industry's Figure A-1 shows a significant change in the slope starting at about 40-45 GWD/MTU. C These data were collected on fuel operated in normal power operations. It is unlikely that this fuel was operated to the extent allowed by technical specifications. The staff has traditionally considered operational transients (not to be confused with accidents) in gap fraction estimates. C Fuel is being operated at ever-aggressive fuel management programs. Two-region cores are in use at some plants, 400-day and longer uninterrupted operating periods are more the norm than when much of the data shown in Figure A-1 were collected. C The data on fuel performance at extended burnups collected from lead test assemblies are possibly limited by the restrictions imposed on the location of these assemblies in the core. Because of these uncertainties, the staff believes it is necessary to maintain conservatism in other aspects of the analyses to compensate. For this reason, the staff cannot accept the industry's proposals regarding radial peaking factor for the fuel handling accident. The staff notes that full core offloads are often performed so that each assembly has an equal probability of being affected by the accident. The NRC staff notes that the ANS-5.4 standard working group has been reconstituted and will meet in April 2000. The ANS-5.4 standard addresses releases from fuel gap during accidents.

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		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	

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		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. *(Attached to this comment package.)*	
8.	Page 12 & 13, 3.2 and 3.3 DG-1081	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 analy-sis. The duration of the gap	As specified on page A-1, the LOCA, like all design basis accidents, is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility. For this reason, the staff did not propose separate source terms for small-break LOCAs. This is consistent with the current standard review plan, which requires that a spectrum of break sizes be used in analyses pursuant to §15.6.5. However, Appendix A of §15.6.5, which addresses radiological analyses, starts with the assumption of the full TID14844 source term. The risk-informing Part 50 initiative may ultimately change the approach to design basis accidents. This regulatory guide would then be revised.

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		release should be consistent with Table 4."	
9.	Page 12, Tables 1 and 2 DG-1081	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."	The NRC staff agrees with the philosophy expressed in this comment. However, the staff believes this information is not necessary for using the data in Table 1 and 2 and belongs in a technical basis document, as it While the final guide is based, in part, on NUREG-1465, the guide avoids addressing NUREG-1465. It is the staff's intent that licensees follow the guide without directly referencing NUREG-1465.
10.	Page 12, Tables 1 and 2 DG-1081	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.	The suggested column has been added.
11.	Page 13, 3.3 DG-1081	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.	The staff agrees that this table needs to be revised. However, the staff believes the appropriate approach is to state 30 secs as the duration of the RCS release phase in PWRs, because (1) this bounds the data given in NUREG-1465 for CE and Westinghouse plants, (2) NUREG/CR-5787, "Timing Analysis of PWR Fuel Pin Failures," (referenced by NUREG-1465) provides support for this value, and (3) no specific value was available for B&W plants. The staff used 2 minutes in lieu of 121 seconds for the BWR to avoid suggesting a high degree of accuracy.
12.	Page 14, 4 DG-1081.	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.	The NRC staff is preparing a technical basis document that will document the bases of all significant positions included in the final guide. This document, when complete, will be available as a public document via the electronic public reading room.
13.	Page 14, 4. 1st paragraph	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 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.	The NRC staff's intent is that these positions apply to any dose calculations performed pursuant to 10 CFR 50.67. Accordingly, the positions would not directly apply to any selective implementation for which dose calculations are not required. However, if dose calculations are required, the TEDE criteria in §50.67 and this section of the guide do apply. The staff has added a statement to clarify this position in the guide.
14.	Page 15, 4.1.5 DG-1081	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."	The affected text in the final guide has been revised as suggested.
15.	Page 17, 4.4 DG-1081	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	While the individual accident appendices provide much of the suggested guidance, the NRC staff agrees that it would be advantageous to detail the guidance in one location, as well as in the appendices. The final guide has been revised to meet the intent of the comment.

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		regulatory guide. Revise Table 6 as follows to add a column on "Dose Duration": Accident - Dose Criteria - Dose Duration LOCA 25 Rem TEDE 30 days unless demonstrated shorter by plant design BWR MSLB 25/2.5 Rem TEDE 2 hrs unless demonstrated shorter by plant design BWR Rod drop 6.25 Rem TEDE 24 hrs unless demonstrated shorter by plant design PWR SGTR 25/2.5 Rem TEDE Until shutdown cooling can remove all decay heat PWR MSLB 25/2.5 Rem TEDE Unaffected SGs : Until shutdown cooling heat removal rate exceeds decay heat Generation Affected SGs : Until primary coolant temperature reaches 212 PWR L 2.5 Rem TEDE Until shutdown cooling can remove all decay heat PWR REA 6.25 Rem TEDE Containment Scenario: 30 days Secondary Side release: Until shutdown cooling can remove all decay heat FHA 6.25 Rem TEDE 2 hrs unless demonstrated shorter by plant design	
16.	Page 18, 5.1.2 DG-1081	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."	The intent of the suggested revision is already addressed in sections 5.1.4 and 5.2.
17.	Page 18, 5.1.3 First Sentence DG-1081	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."	The referenced statement has been revised to address the intent of the comment.

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18.	Page 19, 5.1.4 DG-1081	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."	The NRC staff believes that the assumptions provided in the appendices provided an integrated approach to performing the individual analyses, and generally expects licensees to address each assumption or propose acceptable alternatives which as already noted in section 5.1.4, may be predicated on a previously approved licensing basis consideration. The suggested language reverses the staff's intent. The staff has revised the affected text (C.2) in the final guide to clarify this protocol.
19.	Page 20, 6 DG-1081.	"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 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: C 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. C 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.	This suggestion has merit that warrants further consideration in a venue other than this regulatory guide development. This proposal is independent of the radiological source term and as such is applicable to licensees using either the TID-14844 source term or that addressed in DG-1081. It should be noted that the NRC staff considers the current EQ durations in individual facility design bases to be the result of licensee commitments. Licensees can propose license amendments to change the duration(s) established by these commitments. The duration must be consistent with the period over which the equipment is required to be operable by accident analyses. Recovery actions may be found acceptable if the actions can be shown to be feasible in a post-accident environment.

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		C 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.	
20.	Page A-1, 2. DG-1081	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."	The NRC staff believes that this concept is already addressed in the subject paragraph ("Iodine species for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis…." For clarity, the staff has added "including iodine re-evolution" to the above phrase.
21.	Page A-2, Footnote 1 DG-1081	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.	The staff has converted the footnote to a new paragraph in Position C.3.3. The maximum activity is now defined as the iodine activity in Tables 1 and 2 multiplied by 0.05 for elemental and 0.95 for particulates.
22.	Page A-4, 5.1 DG-1081	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."	The affected text in the final guide has been revised as suggested.
23.	Page A-2, 3.3 DG-1081	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."	The affected text in the final guide has been revised as suggested and as part of the resolution of Comment #21.
24.	Page A-3, 3.8 DG-1081	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.	The intent of position 3.8 is to address the unique configuration of Mark III containment designs in which the technical specification leakrate specified in position 3.7 is insufficient of itself to model the release from the drywell. In Mark I and Mark II containments, the leakrate TS establishes the leakage from the drywell to the secondary containment. However, in a Mark III containment design, the TS leakage applies to the leakage from the containment structure, not the drywell. Position 3.8 provides guidance for modeling the transfer from the drywell to the primary containment. For this reason, the reference to "Mark III" and "primary containment" is intentional. The comment appears to be addressing position 3.5 rather than 3.8. In this regard, all three BWR containment designs have bypass paths through which activity released in the early in-vessel release phase can be released from the reactor vessel without passing through the suppression pool. While the initial RCS blowdown would be to the suppression pool, large portions of the drywell atmosphere could be released without flowing through the suppression pool. For this reason, the staff has worded position 3.5

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			as not allowing suppression pool scrubbing credit as the default, with a provision for case-by-case consideration. The staff recognizes that the current standard review plan section allows some credit for suppression pool scrubbing. However, this was based in part on the conservative assumption that the core release accompanied the RCS blowdown at T=0. In the AST, the core release does not occur at the time of RCS blowdown. To clarify the staff intent, this paragraph is now part of Position C.3.7; subsequent paragraphs have been renumbered.
25.	Page A-3, 3.9 DG-1081	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.	The frequency at which containment purges occur at power does not affect this position. If the purge cannot be isolated prior to the onset of the gap release phase, the radioactivity release needs to be considered regardless of why the purge was initiated or the frequency of such purges. Note that purges related to reducing containment pressure post-accident is addressed in position 7.
26.	Page A-6, 6.3 DG-1081	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."	The affected text in the final guide has been revised as suggested.
27.	A-6, 7. DG-1081	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.	The affected text in the final guide has been revised as suggested.
28.	Page B-1, 1.3 DG-1081	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].	The NRC staff has changed this position in response to comments received and agrees that the activity released from the fuel gap will be CsI. However, the staff has also incorporated text that addresses the expected re-evolution of elemental iodine from the low pH spent fuel pool water due to dissociation of CsI. This is consistent with the conclusion in NUREG-1465 that iodine would re-evolve from solutions with pH less than 7. Ultimately, the isotopic form above the water remains elemental or organic, as originally assumed.

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		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 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	

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		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). 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).	

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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.	Text has been added to the final guide to allow applicants to propose credit for holdup and dilution on a case-by-case basis, consistent with the 2-hour release duration.
30.	Page B-2, Footnote 3: DG-1081	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."	The comment is directed towards those plants that have already been authorized to leave the hatch to be open. However, this appendix also needs to provide guidance to plants that wish to revise their technical specification to allow such operation. The text in the final guide has been revised to address both situations. However, the staff does not agree that credit should be given in radiological analyses for such closure. This is a manual operation that is subject to human performance errors of commission or omission. With few exceptions, the staff does not allow operator actions to replace engineered safeguards features. The language was modified to address plants currently having this option and those that may be proposing this option.
31.	Page E-1 and F-1, 2.1 and 2.2 DG-1081	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.	The references to pre-accident spikes in the draft guide were to the facility technical specification with the parenthetical phrase "typically 60 µCi/gm DE I-131." The licensee may request a change to the technical specification via a license amendment under §50.90. Appendix F allows a multiplier of 335. Appendix E retains the traditional multiplier of 500. These values were developed by the staff in the preparation of the regulatory guidance for alternative repair criteria. This information represents the staff's position based on currently available data.
32.	Page E-2, F-2, H-2 DG-1081	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.	The affected text has been revised to reflect the intent of the suggestion.
33.	Page E-2, 5.5 DG-1081	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."	The staff does not feel that the release mitigation mechanisms, other than those identified in the appendices, have been sufficiently developed to be used as a generic assumption in a design basis analysis. With sufficient technical justification, applicants may propose such credit for staff consideration as an alternative to the assumptions in this guide. The staff does not feel it is necessary to specifically identify this option since it does not believe that such mitigation can be justified with the necessary certainty.
34.	Page E-1, 2.2 (Also applicable to Page F-1, 2.2) DG-1081	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."	The guide has been revised to reflect the intent of the suggestion.
35.	Page E-3, 5.8 (also Page	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	The staff differentiated between SGTRs and technical specification primary-to-secondary leakage in developing this guide. For the SGTR, the draft guide provides the suggested

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	F3, 5.8; page G-2, 5.8; and page H-2, 7.6) DG-1081	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 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."	model in Appendix F for the ruptured steam generator in an SGTR accident. See position 5.6 and Figure F-1. For the intact SG(s) in an SGTR (F.5.7) and for the technical specification primary-to-secondary leakage in PWR MSLBs (E.5.6, 5.7), LRAs (G.5.6, 5.7), and REA (H.7.4, 7.5) the staff position allows licensees to assume no flashing; i.e., the leakage mixes with the bulk water with credit for partitioning if the tubes remain covered. However, if the tube is uncovered, staff positions E.5.8, F.5.8, G.5.8, and H.7.6 called for the licensee to treat the release as flashed without partitioning credit. The staff has revised the guide (E.5.5, E.5.6, F.5.6, G.5.6, and H.7.4) to allow the analyst to use the previous F.5.6 (now E.5.5) model as an alternative.
36.	Page E-2, Footnote 3 DG-1081	Delete this footnote. Primary-to-secondary leak rate is described in item 5.1.	This comment appears to be in error as the subject footnote does not address primary-to-secondary leakrate.
37.	Pages G-1, 2 DG-1081	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.	The affected text in the final guide has been revised as suggested.
38.	Page H-1, 4 DG-1081	Containment Sprays during REA : The following changes are suggested: C Replace the word *LOCA* by *REA*. C 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.	The first suggested change has been made in the final guide. The staff disagrees with the second change. The intent of maintaining sump pH is to control the re-evolution of elemental iodine from the sump. If the sump pH is not maintained, the chemical form of the iodine in the containment will be affected.
39.	Page H-2, 7.3 DG-1081	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.”	The affected text in the final guide has been revised as suggested.
40.	Page I-2, 3 DG-1081	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.	The staff believes that the cross-references to Appendix A provide for appropriate modeling of the transport of radioactivity, and that no additional guidance is necessary.

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41.	Page I-2, 8 DG-1081	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.	The affected text in the final guide has been revised to address the concern.
42.	Page 2, first paragraph of Part B DG-1081	The word "assess-ments" should be "assessments".	This apparent error is an artifact of the downloaded version. The typeset version has this hyphen located at the right margin.
43.	Page 7, 1.3.2 first paragraph DG-1081	Midway through the paragraph there is a sentence that is terminated with two periods. The number of evaluations differs from the first to the second sentence, revise the first sentence to a singular evaluation.	The affected text in the final guide has been revised as suggested.
44.	Page 13, Table 5 DG-1081	The last entry is incorrectly formatted. The columns don't line up.	This apparent error is an artifact of the downloaded version.
45.	Page 17, 4.2.7 DG-1081	The equation should have a reference to the Murphy-Campe report (Reference 20).	The reference has been included.
46.	Page 19, 5.3, Last paragraph DG-1081	In the last sentence there is "?/Q" which should be "?/Q".	This apparent error is an artifact of the downloaded version.
47.	A-3, 3.8: DG-1081	In two places "radioactivity" is written as "radioac-tivity".	This apparent error is an artifact of the downloaded version. The typeset version has this hyphen located at the right margin.
48.	B-1, 2 DG-1081	"decontamination" is written as "decontamina-tion".	This apparent error is an artifact of the downloaded version. The typeset version has this hyphen located at the right margin.
49.	Page E-2, 5.5 DG-1081	The reference should be to a single faulted steam generator, not plural.	This text was removed in the resolution of comment #35.
50.	Page E-1, 2.1 DG-1081	60 Ci/gm should be 60 micro-Ci/gm (comment also applies to page F-1)	This apparent error is an artifact of the downloaded version.
51	Page E- 3, 5.8 DG-1081	The word "need" should be "needs" (this is the fourth word from the end).	The affected text in the final guide has been revised.
52	Page H-1, 4 DG-1081	"4.85&" should be "4.85%".	The affected text in the final guide has been revised as suggested.
53.	Page I-1, 3 DG-1081	Should the reference to "Appendices B through G" be instead to "Appendices B through H"?	The affected text in the final guide has been revised as suggested.
54.	Page 17, 4.2.7 DG-1081	The equation provided, commonly referred to as the Murphy-Campe "geometry factor," is incorrect. The numerator and denominator have been inverted, and the DDE shown in the numerator should be taken outside the numerator. So to correct the formula, you should bring the DDE out, and invert the fraction.	The formula is correct as shown. The Murphy-Campe document noted that "…the dose inside the control room would be substantially less than what the infinite cloud model predicts…." The commenter's formula would increase the estimated dose.
55.	Page 17, 4.2.7 DG-1081	Also related to the above, I do not believe that you mean for the industry to compute a DDE dose value. What you intend, and correctly so, is for us to compute an "effective dose," using the effective dose coefficients from Federal Guidance Report No. 12. There is a difference between DDE (Deep Dose Equivalent) and "effective dose." Not only will the two computations	The NRC staff recognizes that the dose quantities DDE and EDE are, by definition, distinctly different quantities. The NRC staff also recognizes that the EDE would be the optimum dose quantity to use in accident dose calculations. However, the NRC definition of TEDE as specified in Parts 20 and 50 is based on DDE and CEDE. The TEDE dose

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		yield very different results, but they utilize very different technologies. The draft guide does state elsewhere (page 15) that "the DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining TEDE. Federal Guidance Report No. 12 ... provides external EDE conversion factors acceptable to the NRC Staff." There is at least one error in the above cited statement(s), and perhaps two other things that you will want to change. As I stated above, DDE and "effective" dose are not anywhere nearly the same thing, either numerically or qualitatively. The salient reason for a measurement such as DDE is the nature of the dose (I believe, 1cm in depth or 1000 mg/cm2). The idea behind the DDE is that it aligns itself with the fact that dosimetry is used for real and current ALARA dose issues for plant workers. It is not intended to "look" at the real health effects of radiation, and is not a useful expression for hypothetical accident dose calculations. On the other hand, the "effective" dose coefficients, the values of which are found in Federal Guidance Report No. 12, were derived with any entirely different technology. Monte carlo computer codes were used (by Keith Eckerman and his group) to model an anthropomorphic phantom immersed in a semi-inifinite cloud of uniformly contaminated radioactive material. The codes then accounted for the shielding effects of overlying body tissues over various organs, the locations of organs, the dose received by each critical organ, and the weighting factor of each significantly exposed organ, to arrive at the "effective" dose (which is not necessairly the numerical equivalent of "whole body" dose or DDE (which conservatively assumes charged particle equilibrium in the calculations). I believe that you should be allowing, and in fact, requesting, utilities to utilize FGR Nos 11 and 12 for all future calculations. It develops and uses the most sensible and current technology to assess radiation risk. However, you may want to take note of the fact that the "effective" dose coefficient represents a complete change in philosophy of regulation of radiation risks. DDE (Regulatory Guide 8.34) and effective dose (Federal Guidance Report No. 12) are not the same thing. You may want to amend these statements or remove the allusions to the similarity of concepts, and merely state that the NRC Staff accepts the concept of "effective dose" and the coefficients found in FGR No. 12. Second, in terms of the things you may want to modify slightly, the term EDE is not what is presented in FGR No. 12. The term used is "effective" dose coefficient. Third, you may want to note that FGR No. 12 may be used to determine the contribution of external dose to the TEDE, rather than the way it is currently worded, which is: "... EDE may be used in lieu of DDE in determining TEDE," seemingly implying that the only contribution to TEDE is the EDE.	quantity was not defined by NCRP, ICRP, or ICRU, but was developed by the NRC in the development of the Part 20 rule changes in the early 1990's. As explained in the statements of consideration for the proposed and final §50.67, the Commission, in the Parts 50 and 100 rulemaking which became effective on January 10, 1997, deemed that the accident dose criteria should be based on TEDE for all applications for CPs COLs, and DCs after January 10, 1997). TEDE was used in the AP600 design certification rule published in December 1999. Thus, the precedent for the use of TEDE and DDE had been established. The staff recognizes that there have been proposals to revise Part 20 to allow the use of EDE rather than DDE in occupational exposure situations. However, the staff does not believe that it would be prudent to create a definition of TEDE in Part 50 that differs from that in Part 20. It is important to note that the rigorous application of health physics principles and methodologies for dose quantities is not deemed necessary in establishing acceptance criteria for evaluating the performance of plant design features. There is a tendency for people to treat design basis accident analyses as real events when they are simply conservative surrogates for evaluating plant design features. There is no implication that the accident doses postulated would be experienced in probable events or that public or occupational doses at these levels during an accident would be acceptable. The level of conservatism in these analyses and the level of uncertainties are such that the slight uncertainty is introduced by the less-than-optimum use of dose quantities. The staff believes that its position in §4.1.4 regarding the use of EDE DCFs as surrogates for DDE is reasonable and appropriate given the intended use in this regulatory guide. The staff agrees with the intent of the second and third suggested changes. These have been made in the final guide.

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56.	Page B-1, 2 DG-1081	You provide the Spent Fuel Pool DFs of 500 and 1, for elemental for organic species, respectively, and also give the releases as 99.75% and 0.25%. You then compute the overall pool DF to be 200. This final value for pool DF is not correct given the above data. The overall pool DF is computed as follows: DF = 1 / { [Fe/DFe] + [Fo/DFo] }, or DF = 1 / { [0.9975/500] + [0.0025/1] } = 222.469 The formula above is not only consistent with the PWR-GALE methodology (NUREDG-0017), but also must be used to obtain the overall pool DF published by the NRC in Regulatory Guide 1.25. If the NRC Staff desires an overall DF of 200, then the speciation and/or allowable DFs should be modified. If the NRC Staff desires to utilize the existing DFs and speciation, then you should change the overall pool DF to 222.	The NRC staff intentionally rounded the value of 222.469 to 200 in recognition of the uncertainty in the determination of this value. In response to other comments received, the staff has revised the iodine chemical form assumptions for the fuel handling accident. The revised guidance identifies 95% CsI, 4.85% elemental, and 0.15% organic. Because of the low pH in spent fuel pools, the staff assumes that the iodine dissociated from CsI evolves as elemental iodine. Although the organic fraction is lower, the staff has decided not to revise the pool DF assumption further given the uncertainties in the experimental bases of the assumption, the fact that the dissociation of CsI and the re-evolution of elemental iodine may not occur at the pool depth assumed in bubble rise determinations, and uncertainty in conversion of elemental iodine to organic due to impurities in the pool water.
57.	Page A-4, 4.5 DG-1081	As a potential exclusion, on page A-4 you have mentioned that the containment bypass leakage (around secondary containment) should be the value incorporated into Tech Specs. For the plants that do not have such a Tech Spec, the SRP states that a value of 7% should be assumed as annulus bypass leakage. This assumption does not appear to be repeated here. You will perhaps want to restate the assumptions, or otherwise provide the means by which you would review such a facility and how they would perform this work (e.g., adopt such a Tech Spec along with adoption of AST).	The staff found no reference to this 7% value in SRPs 15.6.5, "LOCA," SRP 6.2.3, "Secondary Containment," or BTP CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Designs." In fact, the BTP explicitly requires a plant-specific assessment. SRP 6.2.3 provides for technical specifications on such leakage. The staff believes that that reference to a technical specification value in DG-1081 is appropriately generic. Facilities without such a technical specification may propose an alternative value for the bypass leakage. As 10 CFR 50.36 provides requirements for technical specifications, there is no need for the final guide to provide for adopting a new technical specification.
58.	Pages E-2, F-2, G-1, H-2 DG-1081	Pages E-2, F-2, G-1 and H-2 discuss the effects of Information Notice 97-79. You should note that the draft regulatory guide seems to dictate that this issue be addressed in conversions in the dose calculations, whereas the option is to address this issue within the Chemistry Procedure for determination of PSLR (i.e., include this density correction factor in the Chemistry Procedure in order to adjust the allowable PSLR). In fact, the later method is the method chosen my McGuire and Catawba Nuclear Stations. Before we chose this method, I discussed the issue with Jack Hayes (NRC) and David Steininger (EPRI). They both concurred that our method was both a viable and acceptable method of approach. I authored the dose analysis appendix of the forthcoming EPRI publication, TR-107621, "Steam Generator Tube Integrity Assessment Guideline." In this appendix, I describe the two methods of addressing the issue, and leave it to the discretion of the utility. The only thing not at the discretion of the utility is refusal to address the issue. Hence, this regulatory guide will be out of accord with my guidance, which allows both methods.	The draft guide does not preclude correcting primary-to-secondary leakrate in the leakage monitoring procedure. Paragraph 5.2 only requires that the conversion of volumetric units to mass units consider the density. If the volumetric units have already been adjusted for density changes, then further adjustment is not necessary and is not required by the final guide. Either approach to the correction would be acceptable.
59.	Page H-1 DG-1081	On page H-1, you state that "Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," for acceptable assumptions and methodologies for performing radiological analyses." My only comment here is that you may want to consider the document that I authored for EPRI. I believe that it has much, if not all, of what you need the utility to know concerning radiological calculations for secondary side release accidents (SGTR, MSLB, Locked Rotor and Rod Ejection).	While the EPRI Steam Generator Tube Integrity Assessment Guideline will likely provide useful guidance, EPRI documents are generally not referenced in regulatory documents since most EPRI documents are licensed to EPRI members and are not available to the public. Documents referenced in regulatory guides are generally required to be available to the public. Licensees have the option of proposing use of the EPRI guidance as an alternative to DG-1074.

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60.	Page 20, Note 16 DG-1081	The Tech Spec value being superseded for analysis purposes by values provided in other regulatory guidance. While I understand what you mean, you may want to clarify this statement. The intent is to retain a "margin" between tested values and credited values for purposes of addressing degradation in the nuclear grade charcoal in between surveillance tests. The intent is not to "supersede" Tech Specs, a statement that will confuse many plant workers.	The text in the final guide has been revised to replace “superceded” with “adjusted.”
61	Page 24, Ref.18 DG-1081	You may want to note that this document is "Federal Guidance Report No. 11."	The affected text in the final guide has been revised as suggested.
62	Page 20, 5.1.4, DG-1081	Draft Regulatory Guide DG-1081 mentions several requirements that may not be consistent with a facility's current licensing basis. Examples of these requirements include guidance pertaining to single failure criteria, ESF system leakage rate, and passive failure requirements. These requirements are not directly related to the issue of source term (TID, NUREG 1465, or another Alternate Source Term) and are independent of source term. Assumptions concerning these items should not change and should be consistent with the existing licensing basis of the plant. Section 5.1.4 should be revised to provide clear guidance on these types of assumptions that are not affected by the choice of source terms.	See comment #18 above.
63	Page A-5, 5.3 DG-1081	The guidance contained in the Draft Regulatory Guide specifies the use of conservative values in several phases of the dose calculation. Examples of conservatisms in the dose analysis include the following: The regulatory acceptance criterion for acceptable dose is conservatively low (25 rem TEDE). This limit is designed to and will ensure a very low probability of health effects of concern. The prescribed source term is very conservative compared to the expected source term for design basis accidents (Large LOCA with successful ESF operations). The source term for a design basis accident is expected to be a small fraction of the prescribed source term. The release from the plant is treated conservatively by using the upper limit of containment leakage as compared to expected values. Meteorological parameters of X/Q used in the dose calculation are prescribed to be upper bound values (95%) expected to be encountered only 5% or less of the time as compared to an average value. The dose is calculated for the most limiting receptor at the Exclusion Area Boundary and the Low Population Zone. The use of conservative values and analysis of this type is appropriate. However, there is additional draft guidance that is conservative but unnecessary. Specifically, the requirement for certain plant configurations to assume a passive failure causing 50 gpm leakage for 30 minutes is not mechanistically based and is somewhat arbitrary. The likelihood of having a degraded core scenario represented by this conservative source term evaluation approach in combination	The staff has deleted the passive ESF leakage assumption from the final guide. The staff based this decision on the low probability of this passive failure, especially in conjunction with the probability of a large-break LOCA that results in radioactivity releases of the magnitude assumed in Position C.3 of this guide.

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		with a passive failure is incredibly low. The requirement to assume a passive failure with the resultant prescribed system leakage rate should be deleted from the draft regulatory guide.	
64.	Page F-3, 5.8, Page G-3,5.8, Page H-2, 7.6 DG-1081	These sections address steam generator tube uncovery for short periods of time. Duke Energy considers the effects of steam generator tube bundle uncovery during potential DNB accidents (Rod Ejection, Locked Reactor Coolant Pump Rotor). This consideration addresses the potential for failure of an emergency feedwater pump as the limiting single active failure, where atomization and entrainment of non-flashed primary leakage may occur. Therefore, during periods of tube bundle uncovery for very small leak rate accidents (such as 150 gpd), all primary coolant is assumed to escape without any mixing or depletion of radioactivity. For large leak rates (such as with a steam generator tube rupture), a primary droplet entrainment fraction is computed and applied to periods of tube bundle uncovery.	The NRC staff believes that the referenced sections allow the protocol advocated by this comment. However, the staff has made changes to clarify this situation. See comment #35
65.	Page I-1, 2	This section states that the radiation environment resulting from normal operations should be based on the source term estimates in a facility's Safety Analysis Report or consistent with the facility's Technical Specifications. This section should include a statement or provision that the use of historical data is acceptable when estimating the dose for past operations.	The affected text in the final guide has been revised as suggested.
66.	Page I-2, 4-6	These items provide guidance for estimated doses from the containment atmosphere. The guidance does not address doses due to activity buildup on ventilation filters. Additional guidance concerning buildup on filters is recommended.	The affected text in the final guide has been revised as suggested.
67.	Page I-2, 8	The guidance in this section states that the doses for equipment exposed to sump water should be calculated for a point located on the surface of the water. Shielding codes used by Duke Energy are capable of detailed modeling of the sump, including sump water self-shielding effects, and would be appropriate for these modeling tasks. This section should include a statement that recognizes the use of shielding codes for sump modeling. In addition, when considering sump water in ECCS piping, the pipe size and routing should be incorporated in the analyses to determine the dose rates. The guidance should be revised to reflect these considerations.	The affected text in the final guide has been revised as suggested.
68.	Page 20, 6 DG-1081	The Nuclear Utility Group on Equipment Qualification strongly endorses the concept of establishing accident durations for equipment qualification (10 CFR § 50.49) purposes that are consistent with the accident durations utilized in accident radiological consequences analysis (*i.e.,* for site boundary and control room). Consistent with the provisions of 10 CFR Part 100, the equipment qualification ("EQ") accident duration for loss-of-coolant accident ("LOCA") events should be 30 days. This is termed the mitigation phase in the NEI comments. Issuance of DG-1081 is an opportunity for the NRC Staff to provide unifying guidance for the accident duration used for equipment qualification and radiological consequences purposes. The specific rationale supporting the above comment is discussed in detail below.	See Comment #19.

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69.	Pages 15.0.1-4 Item 3 and 15.0.1-12 Item 5 SRP	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.	The Ci/MWt values used in TACT and RADTRAD are based on values determined for fuel with lower burnup and enrichment. As burnup increases, there is a shift from a uranium fuel economy to a plutonium fuel economy. There will be differences in the production of certain radionuclides (e.g., I-131) due to differences in absorption cross-sections. The buildup in inventory of long-lived radionuclides (e.g., Cs-137) also needs to be considered. The staff has added a note identifying this consideration to Position C.3.1 of the final guide.
70.	Page 15.0.1-6, Table 1 SRP	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": Accident - Dose Criteria - Dose Duration LOCA 25 Rem TEDE 30 days unless demonstrated shorter by plant design BWR MSLB 25/2.5 Rem TEDE 2 hrs unless demonstrated shorter by plant design BWR Rod drop 6.25 Rem TEDE 24 hrs unless demonstrated shorter by plant design PWR SGTR 25/2.5 Rem TEDE Until shutdown cooling can remove all decay heat PWR MSLB 25/2.5 Rem TEDE Unaffected SGs: Until shutdown cooling heat removal rate exceeds decay heat generation Affected SGs : Until primary coolant temperature reaches 212 F PWR LR 2.5 Rem TEDE Until shutdown cooling can remove all decay heat PWR REA 6.25 Rem TEDE Containment Scenario : 30 days Secondary Side release : Until shutdown cooling can remove all decay heat FHA 6.25 Rem TEDE 2 hrs unless demonstrated shorter by plant design	See response to Comment #15
71.	Page 15.0.1-12, III.6.b SRP	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."	See response to Comment #18 above.
72.	Page 15.0.1-13,III.8 SRP	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.	See response to Comment #5
73.	Page 15.0.1-8, III.2.b.(1) SRP	III.2.b.(1), the first sentence reads: "A selective implementation on the basis of only the timing characteristic of an AST will normally	The intent of the first suggested wording has been incorporated in the SRP.

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		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."	
74.	Page A-5, 5.6, 5.7 DG-1081	Informal questions have been received from industry. Some have interpreted the 10% iodine flash to be 10% of the 4.85% projected to be elemental.	The guide has been revised to refer to "10% of the total iodine" (staff's intent), rather than the "10% of iodine" stated in DG-1081.
75.	DG-1081	*Informal question from industry*. For a full scope application where modifications are proposed (such as those which might affect in-containment cleanup systems), the staff position is that EQ evaluations (calculations) should be performed because the cleanup system changes affect the concentration of radionuclides post accident. **Question A:** If the licensee's EQ calculation of record DOES NOT credit removal processes such as containment sprays, filters, or fission product plateout, is it correct to assume the staff position would be to not require EQ evaluation and calculations and to then rely on the generic evaluations performed by the staff? **Question B:** If EQ evaluation and calculations were not required, would the licensee then have the choice s to continued use of EQ based upon TID 14844 or NUREG 1465 ?	Question A: The staff guidance in the DG requires re-analysis only if the proposed change has invalidated an assumption in a prior calculation. In the stated case, no assumption would be invalidated. The licensee could then use the staff's rebaselining conclusions to disposition the AST impacts. Question B: At the present time, TID14844 would be used for re-analysis. However, the resolution of the generic safety issue of the increased Cs in the sump water may change this guidance. Position 1.3.5 was updated to reflect this interim position on the use of TID14844.
76.	Page 4, 1.1.1 DG-1081	*ACRS comment*: The requirement to have prior NRC approval for "changes...that result in a reduction in safety margins" should be reevaluated for removal in light of both the analytical assessments performed by RES and the results of the pilot applications of the alternative source term.	*Staff response to ACRS letter:* The staff will reevaluate this section of the draft regulatory guide during the public comment period. In this re-evaluation The staff will consider the content of the § 50.59 guidance, so far as it is available at that time. The staff considered the 50.59 guidance and the rebaselining study and how they could apply to the final guide. The staff has worked with the personnel developing the 50.59 rule and guidance to ensure consistency. It is important to note that 50.59 does not apply to amendment requests under 50.90 (and 50.67). The staff included this section to remind applicants that safety margins are required to be maintained for modifications that wouldn't be reviewed under 50.59. The guidance allows the licensee to use 50.59 in assessing impact on margins in AST applications subsequent to the initial approved application. While the rebaselining study provided useful insights, the limited sample of plants considered in the study does

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			not provide an a priori basis to summarily disposition all potential plant-specific and modification-specific impacts. Text was added to Position C.1.1.1 to address the use of 10 CFR 50.59 for evaluating subsequent plant modifications once the initial AST implementation is approved.
77.	Various DG-1081	*ACRS comment:* The staff should modify the proposed redefinition of the source term to eliminate the connotation that the release is necessarily to the containment but should retain the wording "...release from the RCS...."	*Staff response to ACRS letter:* The staff does not believe the recommended change to the definition is appropriate. The staff will review the proposed draft guide during the public comment period to ensure that our description of the alternative source term for the LOCA does not misrepresent the NUREG-1465 basis. The applicable section in the final guide is Position C.3.5. The staff has revised this position so that the release for a LOCA is defined as the release from the RCS to the containment. This addresses the ACRS concern. For the non-LOCA accidents, the position states that the same chemical form will be assumed for a fuel handling accident and from the fuel pins through the RCS for DBAs other than LOCA or FHA. The staff recognizes that the definition of source term in 50.2 appears to ignore the role of the RCS in the release of radioactivity from the fuel to the containment. This was a conscious effort on the part of the staff to adapt the NUREG-1465 data to accidents other than the LOCA. The staff understands the ACRS concern that someone using this definition with the tabular data in NUREG-1465 may attempt to credit additional release mitigation in the RCS. However, the final guide provides the only approved alternative source term for currently operating reactors. The tabular data in the final guide are clear in their applicability. While the source terms in the final guide (at least for the LOCA) are derived from those in NUREG-1465, the data were adjusted to address extended burnup fuel and accidents other than the LOCA.
78.	Page I-2, 6 DG-1081	*Informal question received from industry:* All 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." Add to the end: "in cases where such decay is not accounted for.	The intent of the first suggested wording has been incorporated in the final guide.
79.	Page 12, C.3.2 DG-1081	*Informal question received from industry:* Is the gap fraction of I-129 to be assumed as 10%, as specified, or should it be 15% (as for Kr-85), as specified in Reg. Guide 1.25? And what about I-127?	These radionuclides were not explicitly included, but are addressed in the "others" category. The long half-life would suggest a release fraction greater than 10%. However, this is not deemed to be significant given that these iodine isotopes are not significant contributors to dose in reactor accidents.
80.	Page 14, 3.5 DG-1081	*Informal question received from industry:* Section C.3.5 presents the chemical form of the radioiodine releases from melted fuel. However, no reference is made to the composition of gap releases. Appendix B (Fuel Handling Accident) lists this composition as 99.75% elemental and 0.25% organic. Is this the composition to be assumed for all accidents involving gap releases without fuel melt? Section C.3.5 should also be expanded to indicate that the specified composition (with 95% CsI) is only for the airborne source, and to also discuss in general terms the iodine composition for the non-LOCA events.	Clarification was added to Position 3.5. See also response to Comment #28
81.	Page 15, 4.1 DG-1081	*Informal question received from industry:* In Sec. C.4.1 (Offsite Dose Consequences), I believe a new item should be added to indicate that finite-cloud correction (which is essential for elevated releases) may also be applied to ground-level releases. Depending on the receptor distance from the release point, this could lead to a significant reduction in the DDE component of the TEDE.	C.4.1.4 specifies that DDE is based on submergence dose. This is also provided for by the DCF specifications in 4.1.2. This was the staff intent.

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82.	Page 21, 5.3 DG-1081	*Informal question received from industry:* Section C5.3, in addressing the SB doses, states that fumigation should be considered where applicable and timed to be included in the worst 2-hour exposure period. For a BWR with allowed mixing in the reactor building, the release of radioactivity to the atmosphere could peak at a few hours after the postulated accident (sometimes in excess of 8 hrs). I presume one could apply the fumigation condition at that time, but what does one use for the previous time intervals? I believe the fumigation interval determined for the SB dose should be assumed to also apply to the LPZ doses, and to the CR doses as well (if applicable).	The affected section was intended only to clarify that fumigation should be assumed as part of the worst 2-hour EAB dose, and not to exclude fumigation considerations for the LPZ. This was clarified in the final guide. The staff has determined that fumigation has little or no effect on typical control rooms due to the short distances between release points and control room intakes. The elevated plume does not disperse to the ground fast enough to increase control room concentrations. This is reflected in guidance being developed on control room meteorology.
83.	Page C-1, 2 DG-1081	*Informal question received from industry:* Appendix C (BWR CRDA), Sec. 2, presents two RCS concentrations (4 and 0.2 uCi/gm DE I-131). However, only one acceptance criterion is provided in Sec. C.4.3. Should there be two acceptance criteria?	The specification of two iodine sources terms was in error. These have been revised to reflect the guidance in the original SRP section.
84.	Page A-2,3.3 DG-1081	*Informal question received from industry:* If the user opts for the SRP method. What DF applies for the CsI?. The footnote addresses elemental, but not particulate/aerosol.	The suggested text has been added to a new paragraph added to Position 3.3. The footnote has been deleted. See response to Comment #21.
85.	Page D-2, 4.3 DG-1081	*Informal question received from industry:* Reconsider phrase about 2 hours. This should be an instantaneous puff?	Agree. Position 4.3 has been revised to treat release as a puff release.
86.	Page A-6, 6.3 DG-1081	*Internal staff comment:* Add reference to approved GE Topical on MSIV leakage NEDC-31858P	This is approved method for elemental iodine -- conservative for CsI. A cross-reference to the staff SER for the GE topical report has been added to the final guide. The GE topical report is proprietary.
87.	Page C-1, 3 DG-1081	*Informal question received from industry:* As the release from the gap and melted fuel is assumed to be some point beyond the reactor coolant system (like a dry well) these statements can become confusing. Clearly you want the release to go the turbine and condensers in this case. The way it currently reads it almost sounds as if you release to the RCS where it is mixed homogeneously and then some undefined part of that is then released into the turbine condenser area. Perhaps one should add that all the available radionuclides in the RCS, either form the initial conditions of Part 2, or from the gap and melt of Part 3, end up in the turbine-condenser area.	The staff believes that this guidance is already provided. The postulated release pathway for an RDA is via leakage from the MSIVs. The RCS is assumed to be intact. Thus, the drywell is not a factor. Position 3.3 provides for deposition in the reactor vessel steam separators and in the main steam lines. Position 3.4 then addresses deposition in the turbine-condenser.
88.	Page 15, 4 DG-1081	*Informal question received from industry:* Is there a position on decay and daughtering. I talked to a friend a ABB and he thought that you could only count for decay and daughtering in the reactor building.	The decay issue is addressed in the guide. The guide is silent on daughter buildup. Guidance has been added to the final guide that provides that parent-progeny ingrowth should be considered.
89.	Page 16, 4.1.5, DG-1081	Clarifications related to the sliding 2 hour window for dose acceptance criteria The new concept of a sliding 2 hour window to determine the maximum 2 hour dose results in questions that should be resolved in the new guidance. Such issues include: (a) The level of detail in calculated dose information that should be included in license amendment applications or in the Final Safety Analysis Report (FSAR). For example, does the maximum 2 hour dose need to be provided, or should the results for other time periods also be presented to demonstrate that they were evaluated in establishing the maximum 2 hour period? (b) Clarification that a significant increase in the dose during a period that is not part of the worst 2 hour period is not an Increase in Consequences for 10 CFR 50.59 reviews.	Since the regulatory criterion in 50.67 is for the maximum two hour period, the staff would expect an applicant to specify this parameter in the application. The staff would expect to look at results for other time periods only if there were to be some doubt regarding the validity of the two hour period identified by the applicant. This would be handled as a request for additional information. Section 4.1.5 specifies that the maximum dose be reported. The maximum two-hour dose is the parameter considered in determining whether there had been an increase in consequences with regard to the EAB dose, when performing a 50.59 evaluation. If, as a result of the increased dose outside the previous time window, it is necessary to revise the maximum two-hour dose, there would be a basis for evaluating the increase against the 50.59 criteria. However, if the dose for the two hour window did not change, there was no increase in consequences. The staff revised section 4.1.5 to add

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			this clarification.
90.	Page 15, 4.2, DG-1081	Justification that no skin dose limit applies There are current guidelines, such as the SRP acceptance criteria for control room dose, which have an acceptance criterion for skin dose. Total effective dose equivalent (TEDE) does not include the skin dose, and hence the new Alternative Source Term (AST) acceptance criteria contains no limit for skin dose. This difference needs to be discussed in DG-1081. This would facilitate performance of Safety Evaluations for FSAR changes where the current calculated control room skin dose is being eliminated.	The staff addressed the issue of individual organ (capping) doses in the Statements of Consideration for the final 50.67 rulemaking (64 FR 71997 section III.D). In summary, the staff concluded that such limits were not necessary. However, the staff agrees with the need for guidance on performing 50.59 analyses. While there is an industry initiative on guidance in performing 50.59 analyses, the published draft of that guidance does not address the issue of determining the increase in consequences when the "before" and "after" consequences are based on different dose quantities. However, given the difference in publication schedules, the staff has added a footnote to section 1.3.4 that provides guidance in converting the previous whole body and thyroid doses result to TEDE for the purpose of comparison to the new TEDE result when performing a 50.59 analysis.
91.	Page 4, 3, and Page 12, 5, SRP	These two sections state that core inventory should be based on rated thermal power, enrichment and burnup. The codes used in design bases accident (DBA) analyses, such as TACT and RADTRAD, determine core inventory solely as a function of rated thermal power. There are no current requirements, DBA dose models, or guidance on how to adjust the inventory as a function of enrichment or burnup. There is no need to adjust for enrich-ment or burnup as errors associated with these parameters are small compared to the significant uncertainty and general conservatism of other parameters. Reference to these two parameters should be deleted from the SRP.	Please see response to Comment #69.
92.	Page 11, 3 SRP	This section implies that the only radiological aspects to consider in the extension of a containment isolation valve closure time is the timing of the AST gap release for the DBA loss-of-coolant accident (LOCA); There are other accidents such as a control rod ejection where the fuel failure results from the reactivity excursion and hence occurs faster than the DBA LOCA gap release. The containment isolation signal may not occur for this accident until the failed fuel fission products are already dispersed in primary containment. The dose consequences of an increased containment isolation time would have to be addressed for this accident. The importance of addressing other accidents such as the control rod ejection should be mentioned in this section where containment isolation time is discussed.	The referenced section was provided solely as an example in the SRP. See response to Comment #93.
93.	Page 8, 1.3.2 DG-1081	See Comment 2 above in the SRP comments. For a similar reason, this section should also highlight the importance of considering accidents such as the control rod ejection in regard to containment isolation closure time.	While Position 1.3.2 does not specifically identify the accident impacts addressed in the comments, the staff believes that the concern is addressed generically. For example, this section notes that the staff expects that any AST application be supported by evaluations of *all significant radiological and non-radiological impacts of the proposed actions*. The staff does agree a clarification is necessary and has added text to Position 1.3.2. The staff does not believe that the specific example warrants explicit discussion, as Position 3.3 indicates that the release from the fuel gap or pellet is assumed to be instantaneous for non-LOCA DBAs. Therefore, there is no "gap phase."
94.	Page 10, 1.5, DG-1081	The last sentence of the second to last paragraph of Section 1.5 indicates that there are two options for providing the details of the revised radiological analyses in the license amendment application. One is a marked up version of the FSAR, the second is a copy of the actual calculation. The subsequent paragraph seems to provide an acceptable third option. That	The staff intent is that the code inputs would supplement the other submittal information. Submitting the code inputs facilitates the review as the staff will be provided the licensee's analysis model. The correctness of the calculation is pre-determined by the use of the staff-sponsored code. However, the staff still needs to determine with reasonable

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		would be the submittal of the detailed listing of the code inputs if an NRC sponsored code was used (e.g., RADTRAD). The DG should state that this is an acceptable option for the detail needed in the license amendment application.	assurance that the licensee's modeling of the event is appropriate. This may not be readily apparent from the raw code inputs.
95.	Page 12, DG-1081	The DG should specify what the required source term is for an assessment of small break LOCA (SBLOCA) doses. It is assumed that the source term specified in Tables 1 and 2, through the Early In-vessel phase, is for the DBA LOCA which is the large break LOCA. There are numerous licensees who have performed dose assessments for SBLOCA cases, whether it is for environmental qualification (EQ) of components only required for SBLOCA scenarios, or public doses from a release pathway that would onlY exist under SBLOCA conditions. These analyses have typically assumed a release of the gap activity, previously assumed as 10% of the core inventory of noble gas and 10% of the core iodine. It is not clear what should be assumed per the DG for a SBLOCA. Is it the gap release fraction in Tables 1 and 2 or those in Table 3? What is the timing of the release for the SBLOCA, since Table 4 applies to the DBA large break LOCA (LBLOCA)?	The staff has declined to address SBLOCAs in the regulatory guide and SRP. The introductory text to Appendix A notes that a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems, the containment, and the proposed siting of a facility. This is consistent with the definition of a "major accident" in Footnote 1 to 50.67. The NRC staff has never issued a generic source term for a small break LOCA. Section 15.6.5 directs an applicant to consider a spectrum of break sizes in analyzing a LOCA. It directs the applicant to perform a dose calculation using Annex A for the limiting case. Annex A of Section 15.6.5 directs the applicant to assume the Safety Guides 1.3 and 1.4 source term, with no distinction regarding break size. It is understood that some risk insights indicate that the double-ended rupture large break LOCA may not be risk-significant. This will be considered in the effort to risk-inform Part 50. In the interim, the staff will consider the SBLOCA analyses on a case-by-case basis.
96.	Page 12, DG-1081	The titles of Tables 1 and 2 "...Fraction Released into Containment" (taken directly from NUREG-1465) adds confusion on how to treat the release. The new AST rule defines the source term as the "fraction released from the fuel." It is suggested that the titles of Tables 1 and 2 be changed to be consistent with the rule definition of source term. The Appendices then provide specific guidance on where this activity is assumed to be released, whether it would be to the containment atmosphere or the RCS/sump water.	The definition of source term in 50.2 was worded to address LOCAs and non-LOCAs. Tables 1 and 2 address LOCAs, while Table 3 addresses non-LOCAs. The specificity of the titles in Tables 1 and 2 was intentional. The appendices to the regulatory guide do cross-reference the appropriate tables.
97.	Page 12, DG-1081	A note needs to be added that the Early In-vessel column does not represent the cumulative total, but just the fraction released during that phase. For example, the total fraction of halogens released through the Early In-vessel phase at a boiling-water reactor (BWR) is 30%, not 25%. The proper usage may not be obvious to those who have not been involved with NUREG-1465.	See response to Comment #10.
98.	Page. 14, 3.5, DG-1081	This section needs to specify the chemical form in both the gap fraction and the Early In- vessel release. If they are the same, then it should be stated as such. It also needs to be stated as to which accidents these fractions apply LBLOCA, SBLOCA and/or non-LOCA accidents. Technical justifications need to be provided for differences. For example, why is 95% of the gap iodine, cesium iodide (CsI), in Section 3.5, but 99.75% is elemental iodine in Appendix B?	Position 3.5 was revised in response to Comment #77. The staff feels that these changes address the majority of this comment. Technical justifications will be included in a technical basis document that will be prepared in the future.
99.	Page 14, 4 DG-1081	This section needs to clarify the acceptance criteria that will be used for a limited application. For example, a timing-only application uses a combination of the NUREG-1465 timing with the TID-14844 release fractions and only addresses noble gas and iodine. The use of only noble gas and iodine could imply that the whole body and thyroid limits apply. However, since a change in the timing assumption is a change in the source term, it falls under the requirements of 10 CFR 50.67. This rule states that the TEDE criteria must be met.	The criteria of 50.67 apply to all AST applications. Position 1.3.2 provides that if affected design calculations are going to be re-calculated, all affected assumptions and inputs should be updated and all selected characteristics of the AST and the TEDE criteria should be addressed. Thus, if a calculation is performed, then TEDE would be expected and the TEDE criteria stand as stated. If it is a timing only application and no dose calculations are performed, the criteria of 50.67 still apply. The justification not to do a dose calculation is based on the criteria of 50.67.
100.	Page 15, 4.1.5, DG-1081	The last sentence states that the timing increments for dose calculations should be consistent with the rate at which analysis parameters change. The source term allows methods that continuously increase the source term over the phase duration. Hence, the change in one input parameter (source term) is continuous and the time increments could be one second. A	The language of this position was revised in response to Comment #14. The staff believes that the revised language addresses this comment. The staff is reluctant to specify a fixed value for the sliding time window. While the progression of the source term is at a reasonable constant rate, other plant parameters may not be progressing at the

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		statement should be made that a minimum increment for dose determinations of five minutes is acceptable.	same rate. For example, in a steam generator tube rupture, changes in relief valve opening, partitioning, flash fraction, break flow, etc., could change at a rate that wouldn't adequately be captured by a fixed five minute window.
101.	Page 15, 4.1.6, DG-1081	This section should specify that doses beyond 30 days do not need to be assessed for the low population zone (LPZ) dose.	See response to Comment #15.
102.	Page 15, DG-1081	Add a new Section (e.g., Section 4.1.8), and clearly specify that the dose from ground shine from particulate deposition does not have to be considered.	While Positions 4.1.2 and 4.1.4 do specify the dose pathways to be considered, the staff realizes that the reference to Federal Guidance Report 12 could be confusing since this document provides dose conversion factors for contaminated soil. The staff has revised Position 4.1.4 to clarify the cross-reference.
103.	Page 16, 4.2.2, DG-1081	The last phrase, "unless these assumptions would result in non-conservative results for the control room" is contrary to current licensing bases. Control rooms have typically been designed to the DBA LOCA as defined and calculated for the public dose assessment. This is a non-mechanistic scenario with built-in conservatisms to account for mechanistic variations in various assumptions There are numerous examples related to assumptions such as the timing of loss of offsite power, building ventilation flow status, and containment leakage paths that would have to be evaluated to establish the maximum dose. Given the numerous combinations of assumptions, it would be difficult to demonstrate that all combinations of mechanistic assumptions were analyzed. Such requirements are well beyond current rules and practices. Design of the control room to ensure doses remain within 5 REM TEDE for the DBA LOCA used for the public dose analysis simplifies the required analyses and ensures control room habitability for the vast majority of accident scenarios.	The staff believes that the language of the guide is appropriate. The staff believes that the control room habitability doses should be determined using the same Ci/sec release rates that are used for determining the offsite doses. However, there are often simplifying assumptions in offsite dose calculations that may not be appropriate for control room analyses. For example, in offsite calculations, the assumption of the site center for determining X/Q values does not result in significant error at the EAB and LPZ distances. However, the difference in release point location can be significant when considering the substantially shorter distances associated with control room intakes. The staff does not believe that it is beyond current rules and practices for the licensee to identify and assess the credible limiting case. This is fundamental to the deterministic approach to design basis accidents. Positions 5.1.4 and 5.2 address the use of the prior licensing basis.
104.	Page 16, 4.2.5, DG-1081	This section specifies that credit should not be taken for personnel protective devices. This restriction fails to recognize the primary safety aspect of control room habitability which is to prevent the need for evacuation of the control room. Hence, it fails to recognize that the control room would never require evacuation due to iodines or particulates as protective measures, such as respirators or potassium iodide (K') administration, could be used. The only dose that will require control room evacuation is whole body dose, most likely due to noble gas submersion. However, due to the need to meet thyroid or committed effective dose equivalent (CEDE) limits, without credit for protective equipment, licensees typically require designs that minimize iodine or particulate activity, such as a pressurized control room via filtered makeup. Filtered makeup increases the noble gas intake and hence increases the possibility of control room evacuation. Allowing credit for personnel protective measures would simulate the real world, minimize whole body dose, and result in safer control room designs.	The staff position on the use of personal protective equipment in lieu of adequate engineering controls is based on the language of 10 CFR 20.1701 and 20.1702. The staff believes that the use of personal protective equipment (PPE) brings with it increased impacts to the worker's industrial health and safety and, with regard to respirators, decreased effectiveness in contending with the emergency. When it is practical to apply engineering controls, PPE should not be utilized. Potassium iodine is a drug that is controlled by the FDA, which has deemed that use of KI as a radioiodine prophylaxis is indicated only in an emergency in which the health risk of using KI is outweighed by the avoided health risk of the exposure. However, if practical engineering controls can mitigate the radiation exposure, the health risks may no longer be balanced. GDC-19 establishes design criteria, not acceptable doses. In the event of an actual emergency, a licensee would be expected to take actions to maintain exposures ALARA, consistent with the needs of the emergency response. Actions to reduce the thyroid dose at the expense of external exposure are inherently limited by the TEDE design criteria.
105.	Page 18, 5.1.2, and	These two sections could severely restrict the number of licensees willing to submit AST applications and hence preclude the safety benefits and cost saving benefits afforded by more	The staff considers the language of GDC-19, or licensee commitments thereto, to be part of the licensing basis as well, and expects licensees to meet the stated criteria, which are

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	Page 19, 5.1.4, DG-1081	realistic treatment of the source term. There are numerous licensees who have one or more of the following in their licensing basis: (a) Credit for non-safety grade equipment; (b) No assumptions for single failures beyond the base assumption of a loss of one train of emergency equipment or specific agreements that certain single failures need not be considered (e.g., failure of steam generator (SG) blowdown valve to isolate); (c) No consideration of a smart Loss of Offsite Power (LOOP) at the worst possible time for dose mitigation. The LOOP is typically taken at the time of reactor trip as a causal effect. The LOOP at any other time as an independent event further reduces the overall probability of the assumed scenario to a non-credible scenario; (d) Assumptions and methods that are different from current SRP guidelines. (e) The implication of Sections 5.1.2 and 5.1.4 is that the NRC will use the AST submittal to impose changes to these agreed upon criteria in the current licensing basis. The specification of the fraction and chemical form of nuclides released from the core is independent of all of the above considerations. Hence, application of the AST should not require the modification of these current licensing bases.	specified in the context of "any accident, including LOCA." Inherent in this is the expectation that the licensees have evaluated the credible limiting case. If cases with LOOP are limiting, they should be addressed. The staff must make a current finding of acceptability. Additionally, some analysis assumptions have been relaxed. These reductions in over-conservatism need to be considered in the light of existing non-conservatisms. The staff believes that the guidance in the guide and the analysis assumptions in the appendices to the guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations, updated technical analyses or, in some case, a previously approved licensing basis consideration. See also response to Comments #16 and #18
106.	Page 20, 6, DG-1081	This section, combined with Appendix I, tend to imply that a requantification of EQ doses is required. This needs to be corrected and additional guidance added on how the EQ effects can be qualitatively justified. Statements related to the small percent increase in the six month integrated dose due to increased particulates should be included. Guidance should be included on how the licensee can take credit for the inherent conservatisms in current EQ analyses and use these to qualitatively justify the small source term effect. Reference to the generic safety issue on Cesium (Cs) should be made.	Position 6 is a cross-reference to Appendix I, which is guidance for performing analyses deemed necessary. Position 1.3 provides guidance when analyses need updating. Position 1.3.5 provides guidance as to when updating EQ calculations is required. In response to Comments #5 and #75, the staff has revised Positions 1.3.5 and 6 to address the Cs137 issue. Position 1.3.3 provides a example of how known margins in calculations could be used to determine whether a calculation is bounding. No additional changes are deemed necessary.
107.	Page 4, 1.1 DG-1081	This draft Reg. Guide addresses the use of AST to calculate the radiological dose consequences for design modifications. This draft guide should clarify that a use of AST methodology is (or is not) acceptable for facilities pursuing analysis margins in the DBAs without implementing plant physical modifications. Note that Section ffi.3.e.(2) of the Draft Standard Review Plan, SRP-15.0.1, Rev. 0, "Radiological Consequence Analyses Using Alternative Source Terms", discusses NRC review procedures for a licensee proposal of full AST implementation without any plant modifications.	The language of the guide does not preclude applicants from pursuing an AST application without an associated physical facility modification. The guide generally does address applications with associated modifications, as the staff believes that most licensees will not undertake an AST implementation without some expected relaxation in design or operating requirements.
108.	Page 5, 1.1.3 Page 9, 1.3.3 DG-1081	Sections 1.1.3 and 1.3.3 discuss the use of sensitivity analyses (scoping studies) as a substitute to performing detailed design basis accident analyses to show that existing analyses are adequately conservative and re-calculation is not warranted. Additional guidance on how to use this option may be helpful to licensees pursuing this methodology as a complement to a license amendment request. Current draft provides latitudes to show existing analyses bounding but potentially disagreement of approaches could lead to more effort in justifying them.	While the staff recognizes that additional guidance on using sensitivity analyses could be useful, it believes that the guidance provided in Position 1.3.3 is adequate. As experience is gained in the use of this guidance in the context of actual licensee amendment requests, the staff will consider future revisions to this guidance, as necessary.
109.	App. J, DG-1081	In Appendix J, the lower left box (associated with Note 4) seems to indicate that, if scoping analysis indicates TID 14844 is not bounding for all EQ doses, then all EQ doses must be recalculated if AST is implemented. A requirement to recalculate all EQ doses would significantly hinder efforts to implement AST. The contents of the box should be revised to make reference to Section 1.3.3 for guidance on the acceptable use of sensitivity and scoping evaluations, particularly regarding EQ dose calculations associated with implementation of AST.	Changes to Positions 1.3.5 and 6 allow licensees to continue to use the TID14844 source term for EQ calculations.
110.	Page 2, A, DG-1081	In the third last paragraph, replace "This guide is being developed" by "This guide was developed"; and replace "This guide would also identify" by "This guide will also identify".	The intent of the suggested changes has been incorporated.

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111.	Page 2, B, DG-1081	In the first paragraph, replace "to address known uncertainties" by " to compensate for known uncertainties"; and replace "risk assess-ments" by "risk assessments".	The first suggested change has been incorporated. The hyphenation error was an artifact of the downloaded versions only.
112.	Page 4, B, DG-1081	In the last paragraph, replace "the risk4nforming current regulations" by "the impact of risk-informed regulations".	The staff does not feel that the suggested change is correct. The current risk-informing Part 50 effort is not limited to assessing the impacts of such changes, but also pursuing the changes themselves.
113.	Page 7, 1.3.2, DG-1081	In the last paragraph, the example of 25% being a "small fraction" is questionable; perhaps "a small fraction (e.g., 25%)" should be replaced by "a small fraction (e.g., <10%)".	The staff believes that the numerical value of 25% is appropriate. However, noting the potential confusion with "small fraction (10%)" used in assigning dose acceptance criteria, the staff has deleted the word "small."
114.	Page 11, 2.2, DG-1081	Replace "products the into containment" by "products released into the containment"; and replace "the chemical forms of iodine" by "the chemical forms of iodine released".	The intent of the suggested changes has been incorporated.
115.	Page 11, 2.3, DG-1081	"The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events." Either a more elaborated description of how this is accomplished or a reference should be made such as NUREO-1465 if applicable.	The staff believes that the reference to NUREG-1465 in the introductory text of Position 2 provides the suggested guidance. No change is needed to the guide.
116.	Page 14, 3, DG-1081	Replace "the ASTs specified" by "the AST assumptions or parameters specified".	The intent of the suggested change has been incorporated.
117.	Page 14, 3.6, DG-1081	Replace "departure from nuclear boiling" by "departure from nucleate boiling".	The intent of the suggested change has been incorporated.
118.	Page 15, 4, DG-1081	In the first paragraph, replace "dose criteria, and therefore, they do not expect to allow the 'IEDE criteria to be used with TID-14844 calculated results" by "dose criteria; therefore, they do not expect to allow the whole body and thyroid dose criteria to be used with the AST".	The suggested change reverses the staff intent. 10 CFR 50.67 already requires the use of TEDE with the AST. The staff recognizes that much of the guidance in this section could be useful to licensees using TID14844. The referenced statement is intended to limit the use of TEDE to the AST. Under the new 10 CFR 50.59 guidance a licensee can make selected changes in methodology without staff approval as long as the method was approved previously and restrictions and pre-conditions are satisfied. The stated limitation is needed to prevent this generally unacceptable approach.
119.	Page 15, 4.1.4, DG-1081	Replace "calculated using submergence in semi-infinite cloud assumptions with" by "calculated assuming submergence in a semi-infinite cloud with".	The intent of the suggested change has been incorporated.
120.	Page 16, 4.1.5, DG-1081	This paragraph states that the maximum EAB TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a sliding sum over the increments for successive two-hour periods. The range of time for this running two-hour total is not defined, e.g. 8 hours, 24 hours, 30 days. In addition, current emergency preparedness required site evacuation within two hours. So what is the time period of interest?	The intent of the suggested change has been incorporated. The staff does not agree with the statement regarding emergency preparedness as (1) the regulations do not require evacuation within two hours, and (2) the implication that the two-hour window is directly indexed to emergency preparedness is invalid.
121.	Page 16, 4.1.6, DG-1081	Replace "receptor at the outer boundary of the low" by " receptor within the low".	The language used in the guide is consistent with the 10 CFR 50.34, 50.67, and 100.11 provisions on the LPZ. The suggested language would establish a criterion more restrictive than that provided by regulation.
122.	Page 16, 4.1.7, DG-1081	Replace "plume by deposition" by "plume by radioactive decay or deposition"	The staff does not believe that plume depletion by decay is significant for the radionuclides and plume transit times of interest, and therefore, did not specifically include the suggested prohibition.
123.	Page 21, 5.3, DG-1081	Replace "changes in ?/Q analysis" by "changes in x/Q analysis".	This apparent error is an artifact of the downloaded version. The chi character appears in the hardcopy version.
124.	Page A-1, 3.2, DG-1081	Replace "assuming that a 50% plateout of iodine is released from the fuel" by "assuming 50% plateout of iodine released frpm the fuel".	The suggested change has been incorporated.

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125.	Page A-3, 3.7, DG-1081	Replace "but not less than 50% of the maximum" by "but not more than 50% of the maximum".	The intent of the suggested change has been incorporated.
126.	Page A-3, 3.8, DG-1081	Replace "radioac-tivity" by "radioactivity".	This apparent error is an artifact of the downloaded version.
127.	Page A-6,6.2, DG-1081	Replace "but not less than 50% of the maximum" by "but not more than 50% of the maximum".	The intent of the suggested change has been incorporated.
128.	Page B-1,2, DG-1081	In Section 2 of Appendix B, replace "decontamina-tion" by "decontamination".	This apparent error is an artifact of the downloaded version.
129.	Page B-2, DG-1081	The footnotes appear to be out of order.	This apparent error is an artifact of the downloaded version. The footnotes appear in the proper order in the hardcopy version.
130.	Page B-3, 5.4, DG-1081	The footnote "1" is missing.	The footnotes are numbered for each section, not each page. The referenced footnote 1 appears on Page B-2.
131.	Page C-2,3.5 DG-1081	Replace "Such analyses accounts" by "Such analysis accounts".	The suggested change has been incorporated.
132.	Page E-1, F-1, G-1, DG-1081	Change the word "breeched" to "breached" on pages E-1, F-1, and G-1.	The suggested change has been incorporated.
133.	Page F-2, DG-1081	In Footnote 3, replace "Partitioning Coefficients defined" by "Partitioning Coefficient is defined".	This apparent error is an artifact of the downloaded version. The text in the hardcopy is correct. Note that this specific footnote was deleted as part of the response to an earlier comment.
134.	Page H-1, 4, DG-1081	Replace "4.85& elemental iodine" by "4.85% elemental iodine".	The suggested change has been incorporated. See Comment #52
135.	Page H-3, 7.6, DG-1081	A period is missing after the last sentence.	This paragraph was deleted in response to Comment #35.
136.	Page I-2, 4, DG-1081	Replace "neglected since their inclusion would involve a higher degree of complexity than is warranted" by "neglected for modeling simplicity"; and replace "the simpler assumptions" by "the simpler modeling assumptions".	The intent of the suggested changes has been incorporated.
137.	Page I-2, 6, DG-1081	Add the following statement: "This correction factor is not required if the computation from daughter products is included in the analysis."	The intent of the suggested change has been incorporated. See Comment #41.
138.	Page J-1, DG-1081	Note 2 seems to indicate that scoping analyses may be used to some extent for EAB,LPZ,CR, but this is in conflict with Note 3 which precludes scoping analyses for EAB,LPZ,CR.	Note 2 indicates that sensitivity analyses should not be a *significant* part of EAB, LPZ, CR doses. The staff would find that sensitivity analyses used to establish one or more inputs to the overall EAB, LPZ, or CR dose analysis would be acceptable. However the intent is that licensees not use a sensitivity or scoping analysis *in lieu* of calculating doses. A clarification was added.

(Material Provided by NEI in Support of Comment #7)

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.

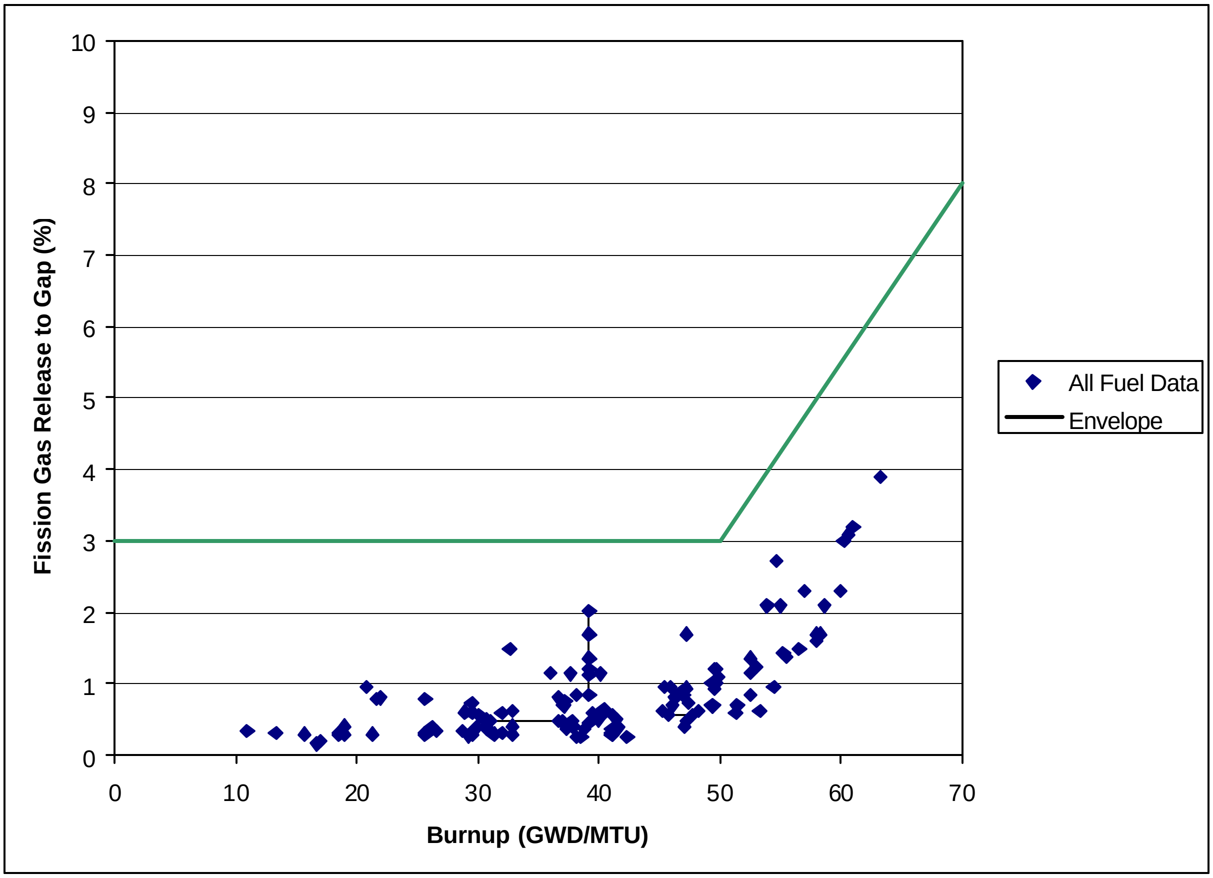
2. G. Smith et al., “Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1,” EPRI report TR-103302-V2, July 1994.

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



COMMENTS TO FEDERAL REGISTER QUESTIONS

The December 23, 1999 issue of the *Federal Register* (pages 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.

STAFF RESPONSE: See deposition table.

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.

STAFF RESPONSE: None required

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, 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 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,

> 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
>
> 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

> the method has been specifically approved by the NRC for the intended plant/application, or
>
> 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

> evaluate selective AST applications approved for other plants to determine if they are appropriate for application at their own plant, and
>
> extend the AST and TEDE criteria to additional technically appropriate applications at their plant

STAFF RESPONSE: The staff has reconsidered this issue. The language of the regulatory guide and SRP was revised to allow licensees to make subsequent modifications based on the selected characteristics incorporated into the design basis by the approved initial implementation under the provisions of 10 CFR 50.59. The revised language states that use of other characteristics of an AST or use of TEDE criteria which are not part of the approved design basis, and changes to previously approved AST characteristics required prior staff approval under 10 CFR 50.67. Thus, a licensee with a timing-only application could, under 10 CFR 50.59, make subsequent modifications that would be based on the timing characteristic that is within the licensee's design basis. However, use of TEDE where it wasn't previously approved, or use of an AST characteristic that is not in the facility's design basis would require prior approval under 10 CFR 50.67. The staff based this

reconsideration, in part, on the revised 10 CFR 50.59 guidance. This revised language follows both suggestions, with the provision that the AST characteristic and TEDE criteria are in the approved design basis for the facility desiring to make the change. This is consistent with the language of 10 CFR 50.67 and the statements of consideration for that rule.

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 report¾that we understand the NRC staff has approved¾to 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.

STAFF RESPONSE: The staff agrees with this conclusion. 10 CFR 50.67 applies to all changes in the source term, which is defined by 50.2 as involving timing. The regulatory guide provides that these timing-only changes may be made as selective implementations without dose calculations.

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.

STAFF RESPONSE: As noted by NEI, it may be possible to re-visit 10 CFR 50.67 at a later date. The staff will evaluate the experience from the 10 CFR 50.59 initiative in considering this option.

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.

STAFF RESPONSE: See deposition table.

<u>SCOPE OF RE-ANALYSES</u>

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.

STAFF RESPONSE: See deposition table.

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.

STAFF RESPONSE: The staff has decided to retain the current language in the regulatory guide. The guide does not require licensees to re-analyze calculations that are shown to be bounding in the initial submittal (with some limitations as provided in Regulatory Position 1.3.) The staff expects licensees to perform analyses using assumptions and inputs that are part of the facility's design basis. Once a licensee has revised its design basis to implement an AST, that design basis source term should be used in all subsequent re-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.

STAFF RESPONSE: See the response to Question 2.A above.

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.

STAFF RESPONSE: The staff recognizes the conclusions of the risk scoping study and expects that these insights will be considered in the risk informing Part 50 initiative. The agency has not concluded that there would be no safety concern relative to the impact of the

increased containment sump water cesium concentration. The resolution of the generic safety issue will determine the safety significance. The staff's conclusion that this situation did not represent an undue threat to public health and safety (warranting plant SD) was based on two factors (1) the public would have already been evacuated for events that could result in EQ doses of the magnitude being considered and (2) the availability of time between the onset of the event and the projected failure provides time for compensatory measures to prevent the equipment failure or restore the degraded safety function.

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*.

STAFF RESPONSE: The staff recognizes the need to consider new requirements against 10 CFR 50.109. This is the reason for initiating the generic safety issue. The staff views the AST implementation as being a means for licensees to voluntarily relax requirements. The staff believes that licensees who voluntarily implement the AST need to address those aspects that are more limiting than the existing guidance as well as those that may be more advantageous economically. The staff does not consider this to be a quid pro quo. This is particularly the case if the change being proposed creates a new issue regarding adequate public protection

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.

STAFF RESPONSE: See the disposition table.

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.

STAFF RESPONSE: Pending the resolution of the generic safety issue, the regulatory guide was revised to allow licensees to use either the AST or the TID-14844 source term in these required re-analyses. This position was included in SECY 99-240, but was inadvertently omitted from the draft guide. This text was added to Regulatory Positions 1.3.5 and 6. Position 1.2.1 of the guide directs the reader to position 1.3 for reanalysis guidance. The staff notes,

however, that the comment ignores the rebaselining study conclusion that EQ doses from components exposed to sump water would increase with the AST.