June 15, 2007

Mr. Randy C. Bunt
Chair, BWR Owners' Group
Southern Nuclear Operating Company
40 Inverness Center Parkway/Bin B057
Birmingham, AL 35242

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: THE BOILING WATER REACTOR OWNERS' GROUP (BWROG) TOPICAL REPORT (TR) NEDO-33148, "SEPARATION OF LOSS OF OFFSITE POWER FROM LARGE BREAK LOCA [LOSS-OF-COOLANT ACCIDENT]," REVISION 2 (TAC NO. MD2917)

Dear Mr. Bunt:

By letter dated August 25, 2006 (Agencywide Documents Access and Management System Accession No. ML062480327), the BWROG submitted for U.S. Nuclear Regulatory Commission (NRC) staff review TR NEDO-33148, "Separation of Loss of Offsite Power from Large Break LOCA." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. Mr. Fred Emerson, BWROG Project Manager, and I agreed that the NRC staff will receive your response to the enclosed Request for Additional Information (RAI) questions by September 28, 2007.

On March 13, 2007, RAI questions were issued, which reflected the additional information required by the Reactor Systems (previously BWR Systems) Branch staff. On May 22, 2007, a telephone conference was held between representatives of the BWROG and the NRC staff. It was agreed upon on that telephone conference that one of the questions from the March 13, 2007, letter needed to be revised. The enclosed RAI questions are the complete set from all reviewing branches, including the revised question from the Reactor Systems Branch staff. If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-1774.

Sincerely,

/RA/

Michelle C. Honcharik, Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 691

Enclosure: RAI questions

cc w/encl: See next page

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ADAMS ACCESSION NO.: ML071630548 **NRR-106**

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REQUEST FOR ADDITIONAL INFORMATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) NEDO-33148, “SEPARATION OF LOSS OF OFFSITE POWER

FROM LARGE BREAK LOCA [LOSS-OF-COOLANT ACCIDENT], REVISION 2"

BOILING WATER REACTOR (BWR) OWNERS' GROUP

PROJECT NO. 691

All page, figure, section, appendix, table, and reference numbers refer to TR NEDO-33148, Revision 2, unless specified otherwise.

A. The following set of request for additional information (RAI) questions are from the Reactor Systems Branch. As mentioned in the cover letter, question 2 has been revised from what was issued on March 13, 2007.

1. The third bullet on Page 4-10 states, "The changes ‘Eliminate LPCI [low pressure coolant injection] Loop-Select’ and ‘One Loop of RHR [residual heat removal] in SPC [suppression pool cooling] Mode’ cannot both be implemented together." Discuss whether it is intended for licensees to be able to implement the RHR in SPC mode change that do not have or have previously eliminated LPCI loop select logic systems.

   Also discuss how, if this change can be implemented in plants without the LPCI loop select logic system, licensees will continue to meet the single failure criterion assuming the availability of offsite power. Provide separate discussions for the following:

   - BWR/3 and BWR/4 that previously had and have eliminated the LPCI loop select logic system,

   - BWR/3 and BWR/4 that were not designed with a LPCI loop select logic system (if any exist), and

   - BWR/5 and BWR/6, which do not have a LPCI loop select logic system.

2. The TR suggests that some plants have already removed the LPCI loop select logic system and that previous modifications to accomplish this task were much more complicated. Describe in detail the process by which a licensee would implement this change under the auspices of this TR and associated exemption requests.

3. The fifth paragraph on Page B.6-1 states:

   > Within a plant type there is variability in the RPV [reactor pressure vessel] size and total ECCS [emergency core cooling system] injection flow. A series of runs was performed using the MAAP4 code to evaluate the impact of a change in RPV liquid volume with the core power level at the 25% uprated condition and

ENCLOSURE

ECCS injection flow unchanged. The base cases for this series of runs were Cases A and D, discussed above. RPV liquid volume, which includes the shroud, lower plenum, shroud head, separators, and active core regions, was varied ±20%. The change in PCT [peak cladding temperature] for these cases was limited to approximately 100 °F for the BWR4 and approximately 200 °F for the BWR6. The largest increase in PCT occurred at lower RPV volumes (refer to Figure B.6-1).

a. From what was ECCS injection flow unchanged?

b. Confirm that the range of liquid volumes analyzed will bound limiting initial conditions from the proposed maximum extended load line limit analysis plus operating state points.

4. How does the thermal-hydraulic analysis account for different fuel designs?

B. The following RAI questions are from the Electrical Engineering Branch.

1. In Section 4.1, the draft NUREG-1829, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process,” report (June 2005) has been recommended for estimating the LOCA frequencies. However, the NUREG-1829 report is still in the draft form. Explain whether a licensee would wait for the finalization of the NUREG-1829 report before submitting changes based on TR NEDO-33148.

2. In Section 4.2, it is stated that the BWROG does not believe that double sequencing events create greater consequences than assumed in this TR. However, the NRC staff has concerns with the postulated double sequencing events. In Section 9 of the NUREG/CR-6538, "Evaluation of LOCA with Delayed Loss Of Offsite Power (LOOP) and LOOP with delayed LOCA Accident Scenarios," the following concerns are stated regarding the treatment of LOCA/LOOP Accidents in the plants' Individual Plant Examination (IPE) submittals:

> The IPEs do not model nor discuss LOCA/LOOP (i.e., LOCA with consequential or delayed LOOP) and the associated Generic Safety Issue (GSI) -171 concerns such as damage to the EDGs [emergency diesel generators] and ECCS pumps, loss of this equipment due to overloading, lockup of a load sequencer, lockout energization of the circuit breakers due to their anti-pump circuits. Some IPEs model the random occurrence of LOOP following LOCA in the LOCA analysis, but these analyses do not address nor provide any insights into the plant's response to the GSI-171 concerns.
>
> The IPEs provide limited information about the protective devices that may be present in a plant to adequately respond to LOCA/LOOP sequences. The information shows that some plants may have some protection against damage to the EDGs and ECCS pumps. Plant-specific information is needed to develop a complete understanding about whether plants have or lack such protective features.

Explain how the licensees will address the above concerns for various LOCA sizes and submit the requisite analyses and the plant-specific information. The submittal should also cover the various concerns discussed in the TR, Appendix D4, "Enclosure 3: Responses to NRC Comments on EPRI [Electric Power Research Institute] Technical Reports 1009110, Revision 1, and 1007966 Regarding the Issue of Double Sequencing Nuclear Plant Safety Loads."

3. In Section 4.4, the "Optimize Emergency Diesel Generator (EDG) Loading" is discussed. Explain whether due to the proposed changes in the loads and the associated sequence, a revised EDG loading profile (for 24 hours) for a worst-case design-basis accident will be developed and submitted for approval. The EDG capability should envelope this loading profile.

   The proposed changes should include any changes to the Technical Specifications relating to the EDG testing. The revised EDG testing should meet the load testing requirement of Regulatory Guide (RG) 1.9, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units as Class 1E Onsite Electric Power Systems at Nuclear Power Plants," Revision 3. Provide any specific exemptions to RG 1.9 with justifications.

4. In Section 4.6, it has been recommended to start EDGs only when needed. This proposed change removes the LOCA start signal from the EDG start logic (EDGs to start only on an actual undervoltage signal). The NRC staff has concerns with this change. The present requirement of the starting an EDG on LOCA signal, even if offsite power is available, provides an added assurance that a redundant onsite power source is readily available and eliminates the EDG starting time in case of a subsequent LOOP. Provide additional justification, if any, (other than already provided in TR) for removing the LOCA start signal from the EDG start logic.

C. The following RAI questions are from the Probabilistic Risk Assessment (PRA) Branch.

<u>Scope of the TR NEDC-33148 and Requested Exemption</u>

1. The TR is intended to support an exemption from Title 10 of the *Code of Federal Regulations* (10 CFR) Sections 50.46(c)(1) and 50.46(d). In a number of places, the TR refers to "the assumption of the simultaneous LOOP with the LOCA" (e.g., bottom of Page 3-1) or "the coincident LOOP and LOCA assumption" (e.g., top of Page 3-2). According to Section 3.1.1, a limited exemption is sought: “The emergency core cooling system (ECCS) design basis would no longer include an assumed LOOP coincident with breaks in ‘large’ pipes - nominal pipe diameters greater than or equal to 10 inches.” Please provide the following clarifications regarding the specific details of the requested exemption request:

   a. In Figure 4-1, the flow chart (step 3) describes a process to determine a LOCA size to be used in an exemption request based solely on the combined LOCA/LOOP frequency being equal to 1E-6/year. Is this the methodology for which the TR is requesting approval, or is a 10-inch break size proposed for all plants with a check that the plant-specific LOCA/LOOP frequency is less than 1E-6/year? Which generic conclusions developed by the TR would no longer be valid if, for example, a plant determined that a much smaller break size would satisfy the 1E-6/year guideline value? For example, the TR states in many places that realistic analysis illustrates that core damage is not expected even with a concurrent LOOP/LOCA. Is this observation a necessary conclusion supporting the proposed exemption? At what LOCA size (i.e., smaller than 10 inches) would this conclusion no longer be valid?

b. Is it the intent of the TR that the exemption only apply to large LOCAs with a "simultaneous" or "coincident" LOOP? If so, would a plant still have to be able to mitigate a large LOCA with consequential LOOP after a slight time delay, rather than "simultaneous" or "coincident?" If the intent is that the exemption be from any LOOP as a consequence of a LOCA greater than a given break size, please state this explicitly in the TR. That is, would the licensing basis analysis for LOCAs greater than some size no longer include any evaluation of the affects of a LOOP on the mitigating systems?

c. It would appear that the intent of the TR is that LOCAs above a certain break size be allowed to assume that offsite power remains available. The exemption would presumably relegate a LOCA above that break size and a consequential LOOP to "beyond design basis." For the large LOCAs with offsite power available, which would remain in the plant's design basis, is it the intent that the single failure requirement of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 35, “Emergency core cooling,” still hold following granting of the exemption?

2. On Page xii, the last sentence in the second bullet reads: "However, an individual licensee may request additional changes on a plant-specific basis as long as adequate justification is provided." The third paragraph in Section 2.4 (Page 2-8) also discusses "other plant changes that are not explicitly described in this report." The TR discusses in some detail seven potential changes that would be made possible if the requested exemption is granted.

   The NRC staff accepts TRs for review in accordance with the Office of Nuclear Reactor Regulation Office Instruction LIC-500, "Processing Requests for Reviews of Topical Reports," which provides guidance for the NRC staff to use in deciding when to accept an TR for review. One LIC-500 criterion is that NRC approval of an TR will increase the efficiency of the review process for applications that reference the TR. Another criterion is that the TR is expected to contain complete and detailed information on the specific subject presented; conceptual or incomplete preliminary information will not be reviewed. The NRC staff notes that allowing changes not described in the TR could be considered "conceptual" or "incomplete."

   Is it the intent of the TR to provide a generic methodology for any change that may be enabled by the granting of the requested exemption, or is the scope of the exemption describe in the TR limited to the seven specific changes identified?

Selection of LOCA Frequencies and Break Size

3. The TR would allow use of NUREG/CR-5750, “Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995,” or NUREG-1829 as a source of LOCA frequencies.

   a. Table 4-1 includes the 5th, 95th, and medium estimates of the LOCA frequencies in addition to the mean value. What is the proposed use of the parameters other than the mean value?

   b. NUREG-1829 was developed specifically to support rulemaking regarding the ECCS requirements. The exemption request proposed in the TR is related to ECCS requirements and, hence, NUREG-1829 would appear to be the applicable reference. Please justify the use of the older, generic NUREG/CR-5750 LOCA frequencies for the exemption request.

   c. NUREG-1829 provides several different frequency estimates for each break size based on the method used to aggregate the expert elicitation results. The NUREG states that the largest sensitivity in the results are based on the modeling assumption about whether the arithmetic mean or the geometric mean of the expert elicitation results are used to developed the LOCA frequency curves. The NUREG describes the geometric mean aggregation as the baseline for the report but states, "the purposes and context of any application must be considered when determining the applicability of any set of study results." As described in Federal Register notice 70 FR 67598, the NRC used NUREG-1829 results to support the efforts to define a new maximum design-basis LOCA. As described in the Federal Register notice, the NRC addressed the modeling uncertainties in the expert elicitation process, in part, by utilizing the results developed by the different methods of aggregating the individual frequency estimates. For example, the quantitative guideline for acceptable LOCA frequency was satisfied by the new maximum design basis LOCA size at the 95th percentile confidence limit of the arithmetic mean value. Please describe the difference between the purpose and context of the decisions sought in this TR report versus the change in the design basis LOCA size described in 70 FR 67598. The differences should justify your proposal to rely on the mean of the geometric mean aggregation technique contrary to the rulemaking, which relied on all the aggregation techniques.

   d. The treatment of uncertainties are a necessary component in all risk-informed decision making. The LOCA frequency estimates include numerous uncertainties including the modeling approximation discussed above. Pages 4-14 and 4-19 of the TR (among others) states that uncertainty must be addressed and documented in the submittal but the discussions seem to relate only to parametric uncertainty in the PRA models. How is the uncertainty in the LOCA frequency included in the sample results reported in Section C.3.8? How is the uncertainty in the aggregation technique modeling assumptions included in the sample results? RG 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003740133), states that one acceptable approach to addressing modeling uncertainty is to perform sensitivity studies to determine the impact of different modeling assumptions (e.g., the aggregation techniques). What guidance will be provided in the TR about how to interpret and reach a conclusion about the acceptability of a proposed exemption request if the results of a sensitivity study indicate that the acceptance guideline is exceeded under some modeling assumptions?

e. NUREG-1829 includes frequency estimates for 60 years as well as for 40 years. Please explain how the use of the 40-year frequency estimates recommended in the TR would apply to a plant that has extended its license to 60 years.

4. The last paragraph on Page 4-3 recommends that, if NUREG-1829 is used, the 7-inch break size be used by licensees in selecting a large break LOCA (LBLOCA) frequency, since it will encompass larger break sizes. It also states that the thermal-hydraulic analyses performed in support of this TR used a 10-inch LOCA break size. Will the requested exemption from 10 CFR 50.46 apply only to LBLOCAs with break sizes greater than or equal to 10 inches? Please explain whether one break size will apply to all BWRs that wish to request an exemption as described in the TR, or does the TR allow for site-specific break sizes? If break sizes other than 10 inches are allowed, would a licensee have to perform plant-specific thermal-hydraulic analyses at the chosen LOCA size?

5. On Page 4-4, second to last paragraph, it says that a licensee should consider updating its PRA LOCA frequencies if they are from earlier sources than NUREG/CR-5750 or NUREG-1829. How would a PRA model using "earlier sources" meet the guidance of RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (ADAMS Accession No. ML070240001), for acceptable technical adequacy, considering this risk-informed exemption request is focused on LOCA events?

6. On Page 2-1, the second paragraph states:

> As an example, a large LOCA with a coincident LOOP could be removed from the design basis if the frequency of that combination of events from all possible contributors (random pipe breaks from all known mechanisms, seismic events, heavy load drops, etc.) were to be assumed to lead directly to core damage, but still meet the framework and RG 1.174 criteria.

In Section 4.1, the methodology allows use of NUREG/CR-5750 or NUREG-1829 LOCA frequency estimates. Please describe how "all possible contributors" to LOCA are addressed in each of these references.

<u>Probability of Consequential LOOP</u>

7. Section 4.2 of the TR is titled: "Determine Plant-Specific LOCA/LOOP Frequency." Table 4-4 provides generic conditional probability of LOOP given a LOCA that were discussed in the attachment to a July 31, 2002, NRC memorandum (Reference 15 in the TR). The first sentence in Section 4.2 states that, "[i]n this step, the plant-specific value is evaluated for the conditional probability of LOOP given a large LOCA." The first

sentence after Table 4-4 states that, "[l]icensees can use the values in Table 4-4 as the source of the probability of LOOP given LBLOCA." These two sentences are contradictory; the first calls for the development of a plant-specific estimate; the second permits the use of the generic estimates. Please clarify whether only a plant-specific method would be acceptable (as implied by the Section 4.2 and Appendix E titles), or whether the TR would allow the option of a generic estimate for the conditional probability of a LOOP given a LOCA (as implied by other parts of Section 4.2 and the example on Page 4-7).

If a generic probability is proposed as an acceptable alternative, please note that the source of the generic estimates (Reference 15) states that "...it is suggested that the generic probabilities of LOOP given a LOCA presented in [Table 4-4] apply only to plants that have certain 'good' characteristics that make them less susceptible to a LOOP given a LOCA. A set of candidate characteristics was developed ...." Section G.5 of Reference 15 also provided preliminary lists of candidate characteristics, classified under the categories "Plant-centered characteristics" and "External characteristics." Since these lists were preliminary, some of the characteristics are difficult to measure. For example, Characteristic 2 under the heading "Plant-centered characteristics" is: "A plant with electrical, instrumentation, and control equipment that is well-designed, analyzed, operated, and maintained ...." This characteristic would be difficult to measure directly. In addition, a description of the way in which a particular licensee would provide sufficient justification for meeting these lists of characteristics was not found in the TR.

Please describe the information that a particular licensee would have to provide to justify that its specific nuclear power plant satisfies these characteristics. Please describe your proposed characteristics in terms of parameters that can be measured and tracked by a licensee and the NRC, and propose suitable performance measurement strategies licensees would be expected to implement to ensure maintaining of these characteristics. The preliminary lists of candidate characteristics described in Section G.5 of Reference 15 may be used as a starting point for preparing your proposed characteristics. The proposed characteristics should cover all the aspects described by these lists.

8. Section 4.2 states that "Section G.4.2 of Reference 15 provides the basis for concluding that the probability of grid-centered events is less than the plant-centered events." The NRC staff notes that the basis for that conclusion may not be valid. For example, the classification of events as "Plant-centered" or "Grid-related" in Reference 15 is based on sparse data. The classification of the events needs to be verified and the conclusion needs to be justified.

A second argument is provided to further support the assertion that grid-centered events are less probable than plant-centered events. This second argument considers all LOOP events, not just consequential LOOP events. It is not clear that data relating to all LOOPs should be used to support conclusions about consequential LOOP events. Further, more recent data presented in NUREG/CR-6890, "Analysis of Station Blackout Risk," states that grid-related performance has worsened in recent years because of 2003 and 2004 data. It further states that future industry performance will indicate whether 2003 and 2004 are outliers or indicative of an increasing trend in grid-related LOOP events.

Please describe the information that a particular licensee would have to provide to justify that the probability of LOOP due to transient factors is less than the probability of LOOP due to plant-centered factors for a given nuclear power plant requesting the subject exemption. Alternately, provide a methodology and justification for calculating a plant-specific probability of consequential LOOP from transient (grid-centered) factors.

9. The third full paragraph on Page 2-7 uses a conditional probability of LOOP given a LOCA of 0.01 in discussing inter-systems LOCA. The NRC staff notes that the conditional probability of a LOOP given LOCA will be a number derived as described in Section 4.2. Also, the generic value for this probability given in Section 4.2 is larger by more than a factor of two. Please explain why 0.01 is used on Page 2-7 when the TR methodology has the licensee determine a specific number for its plant in Section 4.2.

Defense-in-Depth and Safety Margins

10. The forth bullet on Page 3-12 appears to be a statement in response to the third bullet. However, the intent of the defense-in-depth "conditions" is to assess the proposed changes, not the current plant design. Please discuss why the proposed changes do not cause "over-reliance on programmatic activities to compensate for weaknesses in plant design." For example, the proposed change to start one loop of RHR in the SPC mode creates the need for a new operator action for scenarios involving failure of the LPCI train lined up for injection. Please provide an assessment of this and any other new operator actions in terms of potential impact on defense-in-depth.

11. The first paragraph on Page B.1-1 states:

> In order to demonstrate defense-in-depth for the LBLOCA/LOOP separation exemption, per RG 1.174, this Appendix will demonstrate that the LBLOCA/LOOP event can continue to be mitigated, even after implementing the plant modifications discussed in Section 2.4 (LBLOCA/LOOP plant changes).

However, the detailed discussion of some of the proposed plant changes mentions the possibility of core damage for some scenarios. Please clarify the Appendix B statement and answer the following questions.

a. The option in Section 2.4.1 would remove some of the "high capacity pumps needed to rapidly reflood the core following the largest breaks" from the EDG automatic loading. This section also states: "The only potential detrimental effect of this change would be that some LBLOCA events with a concurrent loss-of-offsite power and an additional single failure may lead to core damage." Does this contradict the above statement from Appendix B? Explain. If offsite power is lost and no single failure is assumed, does the analysis in Appendix B demonstrate successful mitigation?

b. The option in Section 2.4.2 would change the initial alignment of one loop of RHR to the SPC mode. This section also states: "a licensee would need to deterministically demonstrate that it could still mitigate the LBLOCA with offsite power available and a single active failure." If offsite power is lost and no single failure is assumed, does the analysis in Appendix B of the TR demonstrate successful mitigation? If offsite power is lost and a single failure is assumed, would that result in core damage?

c. The option in Section 2.4.3 would eliminate the LPCI loop select logic. This section also states: "When offsite power is available, there are no single failures that would prevent reflood following a LBLOCA ...." Does this mean that core damage would be avoided under these circumstances? If offsite power is lost, can the event still be mitigated? Discuss both with and without consideration of a single failure.

d. The option in Section 2.4.4 would allow EDG warm-up prior to loading. This section also states: "Analyses using realistic assumptions, documented in Appendix B, have shown that acceptable PCT values would be maintained for the largest LBLOCA break sizes ...." Does this analysis assume a single failure? If not, would core damage result if a single failure were assumed?

e. The option in Section 2.4.7 would allow increased motor-operated valve (MOV) stroke times. This section also states: "The potential detriment would be that a subset of LBLOCA with LOOP scenarios might lead to core damage." Does this contradict the above statement from Appendix B? Explain. Does this analysis assume a single failure? If yes, would core damage still result if no single failure were assumed?

f. For all options above (a-e), would a licensee be expected to verify that the generic thermal-hydraulic analyses were applicable to its specific plant and the actual plant changes anticipated?

12. Table 2-1 on Page 2-18 shows a change labeled "increased MOV stroke times." The footnote says "nominal valve stroke times ... assumed." What valve stroke times were used for the analysis, nominal or increased? Was any increase in valve stroke time added to the EDG delay? How should a licensee analyze this proposed plant modification in its plant-specific analyses?

<u>Risk Assessment</u>

13. Step 14 in Section 4.15 is to evaluate the risk impact resulting from the combination of plant changes. Does this include the risk of LBLOCA with consequential LOOP? In other words, would the licensee add any risk impact caused by the proposed changes to the 1E-6/year increase in core damage frequency (CDF) (or plant-specific number less than 1E-6) resulting from assuming that a LBLOCA with consequential LOOP cannot be mitigated? What acceptance guidelines would be compared to this total change in risk?

Performance Measurement Strategies

14. Reference 2 is an NRC Staff Requirements Memorandum related to risk-informed changes to ECCS acceptance criteria. In that memorandum, it states that LOCA frequency estimates should be assessed periodically to ensure the basis for decisions made is still valid. Further, RG 1.174 states that an acceptable risk-informed application should include an implementation and monitoring plan to ensure that the conclusion drawn from the engineering evaluation remain valid. The conclusions on the acceptability of the proposed exemption appear to be most dependent on the LOCA frequency and the consequential LOOP probability because changes to these parameters will directly affect the LOOP/LOCA frequency. If a licensee has selected a LOCA size based on the LOOP/LOCA frequence of 1E-6/year, relatively small changes in either the conditional LOOP probability or the LOCA frequency could require that a smaller LOCA size be selected to maintain the validity of the TR. How does the TR propose that the implementation and monitoring monitor whether these parameters are changing and respond to changes that require a smaller LOCA size to be selected in order to maintain the validity of the 1E-6/year LOOP/LOCA guideline?

Miscellaneous Comments and Items for Clarification

- On Page xii, in the first paragraph of the Executive Summary, the NRC staff recommends replacing "govern" with "are incorporated into" in the second sentence, to read: "Today, risk insights, ...are incorporated into the routine operation of power plants ...."

- On Page xiii, in the bulleted paragraph, the first sentence ends: "... which show that there is reasonable certainty that no gross fuel failure will occur and that the reactor coolant pressure boundary, as well as the containment barriers, remain intact, even if the assumed LOOP/LBLOCA were to occur." The LBLOCA and "intact RCS [reactor coolant system] pressure boundary" are contradictory. (Note that on Page 2-5, under "Application of the "defense-in-depth" and "safety margin" philosophies of RG 1.174," the TR correctly states: "In addition, by definition, a LOCA is a breach of the fission product barrier of the reactor coolant pressure boundary.")

- In a number of places in the TR, reference is made to the RG 1.174 risk acceptance guideline of 1E-6 for increase in CDF and/or 1E-7 for increase in large early release frequency (LERF). (See, for example, Pages 2-2, 3-16, and 4-7). However, in Section 4.15, Page 4-18, the second paragraph says to review the risk impact results to see if the RG 1.174 acceptance guidelines are met. The RG 1.174 acceptance guidelines for plants that have a total core damage risk below 1E-4 would be an order of magnitude greater (i.e., 1E-5 for CDF and 1E-6 for LERF.) To be consistent with the rest of the TR and to avoid possible confusion, please make it clear in the second paragraph of Section 4.15 that it is the 1E-6 CDF and 1E-7 LERF acceptance guidelines that are being considered in this TR.

- The risk metrics in RG 1.174 are "acceptance guidelines," not criteria (e.g., TR Pages 2-2 and 2-3). Also, the RG uses "small" and "very small" to describe the risk metrics, but not the word "negligible" (TR Section 4.15).

- On Page 2-4, the numbered list has four items. It appears the first item should be the lead-in sentence, followed by three numbered items.

- Something appears to be missing in the first sentence in the last paragraph on Page 2-4: "Double sequencing would most likely occur if there were a concurrent LOCA, with its associated plant trip, a prior stressed transmission grid condition." Perhaps "concurrent with" should be added after the second comma. Please edit so that the intent of this sentence is clear.

- In the last paragraph on Page 2-5, the third sentence should be changed to make it clear that only LOOP events caused by LOCAs involving breaks greater than a specified size would become "beyond design basis accidents" if the exemption request were granted. Clarify that LBLOCAs below the specified break size would still be included as design basis accidents.

- In the footnote on Page 2-7, it should say "are" vice "is."

- On Page 2-16, the last bullet would be clearer if the ending of the first sentence were changed to read: "... to provide a separate demonstration of defense-in-depth after implementation of the LBLOCA/LOOP exemption."

- Page 3-5, Section 3.1.3.3 states: "For certain BWRs, compliance with the GDC is not part of their licensing and design basis. The following discussion of conformance is not intended to mandate a change to any plant's licensing basis ...." The NRC staff notes that licensees will have to maintain compliance with their plant-specific licensing and design basis, whether that includes the GDCs or not. The TR should clearly set forth the expectation that licensees requesting the subject exemption need to demonstrate compliance with the regulations applicable to their plant in their plant-specific submittal. This thought might be appropriate at the introductory paragraph of Section 3.1.3.

- On Page 3-10, the paragraph just before Section 3.1.4 starts out: "With respect to Item L.3 above ..." It appears this should be "Item L.2," as there is no Item L.3.

- The format in Section 3.2.3 (Pages 3-12 through 3-14) is inconsistent and confusing. The defense-in-depth conditions start out as a bulleted list and change to bold-text headings on Page 3-14. Further, the bulleted list is not consistent with RG 1.174 as stated; some of the bullets are "conditions" and others are supporting thoughts. Recommend making the format of Section 3.2.3 consistent to aid in readability and understanding.

- The discussion regarding defenses against human error on Page 3-14 states, in part, "... these [new actions] are routine operator actions, with reasonable mission times. Thus, the probability of human error is not changed from present." Since these are new operator actions, this statement is confusing. Please clarify whether this statement is meant to discuss the specific human error probability or whether it is a general statement to characterize the new operator actions.

- The last sentence in the first paragraph under Section 4.6 should be removed and detailed guidance provided as to which plant changes would require a quantitative risk assessment.

- On Page 4-12, it is unclear whether the last paragraph is directed at the last bullet on that page, or is more general. It appears to refer to the last bullet; if so, recommend indenting it to improve readability.

- References 3 and 15 appear to be the same, unless for some reason only the transmittal memorandum is meant in Reference 3.

- Reference 18 is to RG 1.200, Revision 1, which is no longer for "trial use," was issued in January 2007.

- Reference 20 is to the American Society of Mechanical Engineers (ASME) PRA standard. Should it include Addendum B, ASME RA-Sb-2005, dated December 20, 2005?

- On Page 4-7, the last two sentences of the second paragraph in Section 4.3 are confusing:

  "The second option ... select one of the LBLOCA frequencies described in step 2 ..." Step 2 is the determination of consequential LOOP frequency, not LBLOCA.

  "If the second option is chosen, select the LBLOCA frequency that is most applicable to licensee's plant and PRA model ...." It is not clear what is meant by "applicable to ... PRA model."

  In both sentences, should the consideration be of the frequency of LBLOCA and consequential LOOP? Please clarify what is meant by these sentences and suggest revisions as needed.

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