Response to Enclosure to NRC RAIs

Containment and Ventilation Branch General Comments (page 1-2)

1. Section 9.1, lines 363-366 state, "Consistent with standard scheduling practices for Technical Specifications Required Surveillances, intervals for recommended Type A testing given in this section may be extended by up to 15 months. This option should be used only in cases where refueling schedules have been changed to accommodate other factors." The NRC staff feels that this passage, unchanged from Revision 0, needs to be revisited.

With the test interval at 10 years, the NRC staff accepted this passage, seeing it as a "last resort" when some unexpected delay in starting a planned refueling outage pushed it out beyond 10 years. However, experience indicates that the wording of the last sentence is not restrictive enough to keep licensees from tacking on the 15 months whenever they want. Conventional wisdom is that most licensees simply think of the test interval as 11 years and 3 months and plan accordingly from the beginning of a test interval. This is a different industry interpretation of the sentence than the NRC staff intended.

With the test interval increased to 15 years, the original wording is no longer acceptable. It should be changed to shorten the "leeway"period from 15 months, to between 6 to 9 months, with a basis provided. A comparable revision to the "leeway" period should also be made to Section 11.3.

Response to general comment 1, page 1:

The last paragraph in Section 9.1 of NEI 94-01 will be changed to read as follows:

> ~~Consistent with standard scheduling practices for Technical Specifications~~ Required surveillances, intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions, but should not be used for routine planning and scheduling purposes. ~~This option should be used only in cases where refueling schedules have been changed to accommodate other factors.~~

The last sentence in Section 11.3 will be deleted entirely.

2. Section 9.2.3.2: One stated objective of this revision is to incorporate into it the exceptions cited in Regulatory Guide (RG) 1.163, so that NEI 94-01 will be acceptable on its own. Exception C.3. of the RG states that visual examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years.

Section 9.2.3.2 states that the examinations must be conducted prior to each Type A test and at periodic intervals between Type A tests as specified by the applicable year and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL. The NRC staff request a discussion of whether, and how, this provision is consistent with exception C.3. of the RG, considering especially the longer 15 year interval.

Response to general comment 2, page 1:

The language in Section 9.2.3.2 will be changed to read as follows:

> To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and ~~at periodic intervals between Type A tests as specified by the applicable year and addenda of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL~~ *during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years."*

Containment and Ventilation Branch Editorial comments/typographical errors (page 2)

1. Section 1.2, line 84 states, "...reducing the frequency of Type A tests ([integrated leak rate test] IRLTs) from the current 3 per 10 years to 1 per 15 years...." Considering that no plant does 3 tests in 10 years anymore, delete the words "the current."

Response to editorial comment 1, page 2:

To obtain results from the risk impact assessment that would be comparable with previous work (e.g., NUREG-1493 and EPRI TR-104285) that used 3 tests per 10 years, this was chosen as the base for the latter risk impact assessment. The words "current 3 per 10 years" will be replaced with *"baseline (3 per 10 years)".*

2. Section 6.0, line 237 states, "The installed isolation valve seal-water system fluid inventory is sufficient to assume the sealing function for at least 30 days at a pressure of 1.10 Pa." The staff believes that the word "assume" should be "assure."

Response to editorial comment 2, page 2:

The word "assume" will be changed to *"assure".*

3. Various locations: American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-1994 is cited in numerous locations, but in two different formats. Sometimes there is a hyphen between "ANS" and "56.8," and sometimes it appears without the hyphen. Please be consistent.

Response to editorial comment 3, page 2:

The convention: *ANSI/ANS-56.8-2002* will be used, with hyphens.

4. Section 10.2.2.1, line 735: The term "Pac" has not been changed to "Pa".

Response to editorial comment 4, page 2:

P_{ac} will be changed to P_a.

5. Section 11.2, line 894: Capitalize "type A."

Response to editorial comment 5, page 2:

The "T" in Type A will be capitalized.

6. Section 11.3.2, line 1132 states, "...under Option B to 10 Code of Federal Regulations (CFR) 50, including...." It should mention also Appendix J, as in "...under Option B of Appendix J to 10 CFR 50, including...."

Response to editorial comment 6, page 2:

The reference to the regulation will be changed to read: …"*test intervals under 10CFR50, Appendix J, Option B*, including plant-specific"…

Geosciences and Civil Engineering Branch General Comments (pages 3-4)

1. Executive Summary: In the third paragraph, the Revision 0 provision of performing a Type A test after identifying the cause and instituting corrective action has been deleted in this revision. The only way to identify the leakage characteristics of the containment after corrective actions is to perform a Type A test. Please provide justification for this deletion.

Response to comment 1, page 3:

'Demonstration of an acceptable performance leak rate' in the context given was intended to require performance of a Type A test. For clarity, the words "*during a subsequent Type A test"* will be added to the paragraph as follows:

If the Type A performance leakage rate is not acceptable, the performance criterion is not met, and a determination should be performed to identify the cause of unacceptable performance and determine appropriate corrective actions. Once completed, acceptable performance should be reestablished by demonstrating an acceptable performance leakage rate *during a subsequent Type A test* before resuming operation and by performing another successful Type A test within 48 months following the unsuccessful Type A test. Following these successful Type A tests, the surveillance frequency may be returned to at least once per 15 years.

This change will also be made to Section 9.2.6.

2. Section 1.1, line 13: The NRC staff notes that you use the 1994 version of ANSI/ANS-56.8 (the Standard). The 2002 Edition of the Standard utilizes performance based criteria for the containment leakage rate tests. Provide the basis for not using the most recent edition of the Standard. In addition, for consistency and accuracy, direct references to the provisions of the Standard, where applicable and acceptable, should be made, rather than paraphrasing.

Response to comment 2, page 3:

NEI 94-01 will be revised to refer only to *ANSI/ANS-56.8-2002.*

3. Section 1.1, lines 32 to 45: The fact that Nuclear Regulatory Commission Technical Report (NUREG) -1493 arrives at a statement of "imperceptible increase in risk" is based on considering non-degraded and ideal containments. It did not consider the realistic containment vulnerabilities,

and the explicit criteria for risk-assessment were not available at that time. In spite of all the efforts to relate ILRT interval to risk parameters, it appears that the risk parameters considered are insensitive to the ILRT interval. In reality, the containment-components of operating reactors are degrading, and pragmatic considerations would require an assessment of overall integrity (leakage rate) of the containment, as a minimum, every 15 years. The NRC staff requests a discussion, in the appropriate sections, which provides guidance to address current containment conditions.

Response to comment 3, page 3:

The following language is to be added at the end of Section 1.1:

….assessments are required. *Moreover, pragmatic considerations require an assessment of the overall integrity of the containment, including Type A integrated leak rated testing at fifteen-year intervals.*

4. Section 1.1, lines 52 to 58: If the exemptions were issued after the Technical Specifications (TS) were approved, when the licensee amends the TS requirements to the new test interval (for Type A, Type B, or Type C tests), it should explicitly describe which exemptions the licensee wants to continue with and which exemptions it will not use during the implementation of the new test intervals. This information should be part of the TS amendment request. The NRC staff requests that this section be clarified to state that this approach is acceptable provided the NRC has a chance to review the licensee's choice, as part of the TS amendment.

Response to comment 4, page 3:

This paragraph in Section 1.1 has not changed from NEI 94-01, Rev. 0. The language was provided to clarify to plants that previously established exemptions would not be invalidated by implementing Option B of 10CFR50, Appendix J.

Industry agrees that identification of those exemptions which will be in force, and those which will not be used in the future should be provided in the TS amendment request. A Technical Specifications Task Force (TSTF) Traveler will be developed and submitted to the NRC when NEI 94-01, Revision 1 is approved and issued. This Traveler will be the model TS to be followed by plants and will include this requirement.

The following sentence will be added to the end of the second to last paragraph of Section 1.1:

However, any exemptions to provisions of 10CFR50, Appendix J to be maintained in force as part of the Containment Leakage Testing Program should be clearly identified as part of the plant's program documentation.

5. Section 3.0, lines 145 to 148: This provision should apply to (1) the plants which do not want to extend their ILRT interval beyond 10 years, and (2) the plants which do not want extend their ILRT interval beyond the one-time 15 year extension. In the second case, the plants will have to revert to a 10 year interval.

Response to comment 5, page 3:

The last paragraph will be changed to: *…and who do not wish to extend ILRT surveillance intervals beyond ten years, including ten years with a one-time extension of the interval up to fifteen years are not….*

A similar change will also be made to Section 1.1.

6. Section 6.0, lines 194 to 200: Irrespective of the impact of the design leakage rate on risk, General Design Criterion 16 states, "Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity ------." The purpose of the overall leakage rate test (i.e. Type A test) is to verify that the containment retains its essentially leaktight condition. La is a surrogate for an essentially leaktight condition. This type of discussion is appropriate in these lines.

Response to comment 6, page 3-4:

The language contained in the first paragraph of Section 6 is a direct quote from 10CFR50, Appendix J, Option B. It is provided at this place in NEI 94-01, Revision 1 to articulate the requirements of Option B, and to establish the context of the following paragraphs in this document. A direct quote from the regulation was considered more appropriate for NEI 94-01, Revision 1 vs. the paraphrase in NEI 94-01, Revision 0.

7. Section 6.0, lines 215 to 221: For the sake of completion and consistency, it is suggested that the provisions of Sections 6.4.4 and 6.5 of the Standard (ANSI/ANS-56.8-2002) be provided in a few paragraphs in this area. Periodic revision of the administrative limits based on operating experience should be emphasized.

Response to comment 7, page 4:

NEI 94-01 will be revised to refer only to *ANSI/ANS-56.8-2002.*

8. Section 8.0, lines 259 to 275: Section 3.2.5 of ANSI/ANS-56.8-2002 has the performance-based guidelines and envelopes the provision in the four bullets. For consistency with the referenced documents, the staff suggests that instead of repeating and abbreviating the Standard's provisions, this NEI report should reference the Standard for draining and venting requirements. In general, this Section has a lot of redundancies with the Standard, and the provisions in this report should point out additional practical guidelines without repeating the content of the Standard.

Response to comment 8, page 4:

NEI 94-01 will be revised to refer only to *ANSI/ANS-56.8-2002.*

9. Section 9.2.2, lines 453 to 458 state, "The interval for testing should begin at initial reactor operation," which contradicts the earlier sentence, "The first periodic Type A test shall be performed within 48 months after the successful completion of the last preoperational Type A test." The staff agrees with the earlier sentence on lines 475-476.

Response to comment 9, page 4:

The second paragraph will be changed to read as follows (*insertions in italics)*:

The first periodic Type A test shall be performed *after commencing reactor operation and* within 48 months after the successful completion of the last preoperational Type A test. Periodic Type A tests shall be performed at a frequency of at least once per 48 months, until acceptable performance is established in accordance with Section 9.2.3. ~~The interval for testing should begin at initial reactor operation.~~ Each test interval begins upon completion of a Type A test and ends at the start of the next test.

10. Section 9.2.3.3: To ensure that licensee risk-informed assessments are of sufficient quality, the NRC staff requests that NEI propose an approach to ensure that Type A leak rate test results from industry operational experience data are monitored. As appropriate, this data should be utilized in plant-specific ILRT assessments to demonstrate that risk acceptance guidelines reflect insights from the most current data regarding containment degradation. As new information becomes available, after fifteen year ILRT implementation, licensees should periodically reevaluate this conclusion.

Response to comment 10, page 4:

The analysis is intended as a one-time risk assessment to confirm the plant-specific acceptability of extending the ILRT interval. Existing programs in both industry and NRC are in place to collect and disseminate operating experience. Any significant containment degradation or ILRT failure with potential generic implications would be noted in the plant's operating experience review and appropriate action taken.

11. Section 9.2.4: With an ILRT interval of 15 years, the deferral from the Type A test provided in this Section is inappropriate. At this time, the NRC is providing relief from performing ILRT after SG/RPV or penetration replacement and requiring licensees to perform short duration structural tests to get an assurance of compatible modification.

Section 9.2.4 will be changed to read as follows:

Repairs and modifications that affect the containment leakage integrity require *local* leakage rate testing *or short duration structural tests as appropriate to provide assurance of containment integrity following the modification or repair. This testing shall be performed* (~~Type A testing or local leakage rate testing~~) prior to returning the containment to operation. ~~Testing may be deferred to the next regularly scheduled Type A test for the following repairs or modifications:~~

- ~~Welds of attachments to the surface of steel pressure retaining boundary;~~
- ~~Repair cavities, the depth that does not penetrate required design steel wall by more than 10%, or~~
- ~~Welds attaching to steel pressure retaining boundary penetrations where the nominal diameter of the welds or penetrations do not exceed one inch.~~

12. Section 10.2.3, lines 771-780: From a practical point of view, the initial testing of the valves should be performed at every outage until a plant specific performance history is developed for each of the valves.

Response to comment 12, page 4:

This section of NEI 94-01 has not been changed from that promulgated in NEI 94-01, Revision 0. Section 10.2.3.1 requires performance of Type C tests at a frequency of at least once per 30 months until adequate performance has been demonstrated. 30 months is not inconsistent with the Appendix J, Option A maximum interval of 24 months, + 25%.

Probabilistic Risk Assessment Licensing Branch A General Comments (pages 5-8)

1. The scope of the EPRI report and methodology is limited to extension of the Type A interval. Several boiling-water reactor (BWR) Mark III utilities have applied a similar methodology to support extension of the drywell bypass test (DWBT) interval. The EPRI report does not address the DWBT test interval. Clarification to this effect should be provided within the document, and to the risk impact assessment template.

   Response to comment 1, page 5:

   Yes. The scope of the EPRI is limited to the risk assessment of the extension of ILRT intervals. The report will clarify that the risk assessment of the extension of ILRT intervals can be affected by other risk informed applications that resulted in changes to testing or maintenance intervals. Of particular interest in the calculation of the risk associated with the ILRT are those risk informed applications that result in changes to the containment testing or maintenance. These changes, if any, should be accounted for in the assessment of the risk associated with the extension of the ILRT interval.

2. Section 2.1, the 1st paragraph states, "the risk impact assessment will generically assess the risk impact . . ." This statement appears to oversell the assessment, since it is largely limited to two example applications, does not reflect on or attempt to draw generic conclusions from the previous evaluations summarized in Appendix G and ultimately calls for plant-specific, confirmatory risk assessments, thereby contradicting the claim of a generic assessment.

   Response to comment 2, page 5:

   The last sentence of the 1st paragraph will be changed to read: …impact assessment will assess the risk impact of the 15 year testing interval in a *generally conservative* manner and consider industry experience and appropriate regulatory guidance (RG1.174)[4]. *Two conservative, (but not bounding) example plants are considered in this risk impact assessment. A template for the individual plant risk impact assessments required by NEI 94-01R1 is contained in this report as Appendix H.*

3. Section 2.1, next to last paragraph, and Section 4.2.1 (also applicable to pages H-9 and H12): The NEI Interim Guidance is actually contained in two NEI letters – a November 13, 2001, letter that provides interim guidance, and a November 30, 2001, letter that provides additional information. Both letters should be cited.

   Response to comment 3, page 5:

   Both letters will be cited.

4. Section 4.2.2: The NRC recommends (1) mentioning that the consequence analyses performed as part of the Severe Accident Mitigation Alternative (SAMA) analysis for license renewal is one source of plant-specific population dose information, and (2) clarifying that site-specific dose information from either the plant-specific probabilistic risk assessment (PRA) or SAMA analysis, or the scaling of reference plant population doses (as described in Sections 4.2.2, 5.1.2, and 5.2.2) should be used, rather than the generic population dose values from the NEI Interim Guidance (which some licensees have used directly).

Response to comment 4, page 5:

Yes. The guidance will be clarified to emphasis the use of the most relevant plant specific information should be used for population dose information. The order of preference shall be plant specific best estimate, Severe Accident Mitigation Alternative (SAMA) for license renewal, and scaling of the reference plant population dose.

5. Section 4.2.2: Adjustments to reference plant population doses to account for differences in containment allowable leakage rates are reasonable, but further adjustments to account for differences in containment free volume are unnecessary, since the relationship between containment leak area and free volume are already captured by expressing the containment leakage rate in terms of volume percent per day.

   Response to comment 5, page 5:

   The report will be modified to clarify that adjustment based on containment free volume should not be made.

6. Section 4.2.3: Recommend adding a discussion regarding the levels of risk increase (population dose) that are considered small. This should be addressed in terms of both percentage increase and absolute increase (i.e., person-rem per year), and tied back to the conclusions in NUREG-1493 and the results from the approximately 50 integrated leak rate test (ILRT) submittals prepared to date. (This comment also applies to Sections 5.1.3 and 5.2.3.)

   Response to comment 6, page 5:

   Based on a review of the available approved ILRT submittals and NUREG-1493, the population doses considered small will be defined.

7. Section 4.2.5: Recommend providing a description of the corrosion events identified to date, and the applicability of these events to various containment types/regions. For the example applications in Sections 5.1.5.1 and 5.2.5.1, should provide the basis for the assumption that only two of the observed failures are considered applicable for the example plants.

   Response to comment 7, page 6:

   The corrosion events are described briefly in appendix A. Additional description of the two failures considered will be provided in the report. The two corrosion events referred to in Sections 5.1.5.1 and 5.2.5.1 are the backside thru-liner penetration events that occurred at North Anna and Brunswick. As of the publication date for this report, there had been no other similar reported events resulting in thru-wall penetration of containment liner due to corrosion. The sample sensitivity assessment guidance provided in Sections 5.1.5.1, 5.2.5.1, and Appendix H, Section 4.4 considers potential leakage pathways in the vicinity of the containment cylinder and dome and the basemat. This material is based on the Calvert Cliffs analysis approved for many plant one-time ILRT interval extensions.

8. Section 4.2.6, 1st paragraph: Although in concept a large pre-existing leak could preclude late containment over-pressure failure and consequential core damage in "TW" sequences, such scenarios could still lead to core damage if the leakage location leads to a hostile environment (e.g., high temperature or flooding) in the vicinity of the emergency core cooling system (ECCS) pumps, or if the leakage magnitude is not sufficient to relieve gradual over-

pressurization (e.g., if it is marginally greater than 35 La.) Taking credit for a pre-existing leak is non-conservative and an unnecessary complication in the methodology and should not be suggested.

Response to comment 8, page 6:

While it is realized that there is a remote possibility that the postulated leak size could be a specific size that resulted in gradual pressurization of containment, the resulting dose rates would be small (<35La) and therefore not large and early. Also, it is realized that the leak could be in the specific location that failed injection sources (although TW is largely a BWR issue and sources could be in the turbine building (feedwater or condensate) or other buildings (fire protection). However, this is considered very remote.

In any event, the suggestion to consider the reduction in TW sequences will be removed from the methodology and report.

9. Section 4.2.7 (also applicable to pages H-8 and H-43): The document sets too low an expectation regarding consideration of external events, by deferring this topic to a section labeled "other considerations" (almost as an afterthought), and by stating that in cases where the increase in large early release frequency (LERF) is less than 1E-7 per year the contribution of external events can be addressed qualitatively. Recommend that the document call for a quantitative assessment of the contribution of external events, to the extent supported by the licensee's external event risk models. If the licensee's risk models include fire and seismic PRAs it is reasonable to expect that external events (and impacts on LERF and ΔLERF) would be treated quantitatively. Even when the risk models are based on margins or screening approaches, some degree of quantification (based on simplifying assumptions) is reasonable.

Response to comment 9, page 6:

It is agreed that based on current expectations the document sets too low of an expectation with regards to external events. It is proposed that the External Events topic will be given a methodology and report section. It will be further recommended that to the extent that the analyses are available they will be used to quantitatively assess the risk impact of the ILRT interval extension. In addition, some degree of quantitative assessment will be suggested when margins or screen approaches were used in the assessment of external events. However, should the only valid models for external events be qualitative in nature or the quantitative models are not of sufficient quality, an assessment will be performed based on analyst judgment.

10. Sections 4.2, 5.1, and 5.2: The methodology discussion and both of the example applications are silent on the issue of containment over-pressure, and whether a large leak could result in a potential increase in core damage frequency (CDF) for the example plant. This issue should be addressed as part of the methodology and example applications. Licensees need to verify that credit for over-pressure is not required to assure adequate ECCS operation, or perform a plant-specific assessment to supplement the evaluation called out in the topical report. The methodology should indicate that a traditional license amendment request should be submitted for those plants that require containment over-pressure for adequate ECCS net positive suction head.

Response to comment 10, page 6:

The methodology will be revised to emphasize that if containment overpressure is required for ECCS pump operation that the affects of loss of ECCS given a containment leakage event must be reflected in the risk evaluation. In addition, the CLLIP notice will indicate that the CLLIP only applies to those plants that do not credit overpressure for ECCS pump operation and a traditional license amendment require is required in these cases.

11. Section 5.1: The Vogtle assessment is atypical in several regards, calling into question whether this is a good example for the pressurized-water reactor (PWR) application. Some of these aspects are: (1) a very high fraction of the core damage frequency (CDF) assigned to the intact containment class (.994), (2) a total release frequency which is less than the total CDF, necessitating scaling the release frequencies to match the CDF (in this example, the same scaling factor of 1.116 was applied to all release classes without justification), (3) a lack of information on seismic risk, (4) only a limited assessment of external events, which considers only the impact (of including external events) on total LERF rather than the impact on both the risk increase and the total risk. Each of these aspects should be further addressed in the report if this plant analysis is retained as the example application.

Response to comment 11, page 6:

Yes, the Vogtle plant is an atypical case given the very large fraction of core damage frequency that results in an intact containment. This plant was chosen for several reasons. The first is that the Vogtle ILRT submittal was approved, recent, contained treatment of external event and treatment of corrosion events. Also, with the atypical high intact frequency, the Votgle plant bounds most other PWRs since a high intact containment results in a high delta large early release frequency. In the case of the second point, this is indeed atypical but provides a basis for consideration of plant specific and PRA specific treatment. The lack of information on seismic is not atypical since significantly less than half of the US plants performed any seismic quantitative PRAs. Additional information can be provided in the report to inform the reader of some of the plant and PRA specific aspects of the example application.

12. Sections 5.1.5.1 and 5.2.5.1: Recommend adding a summary statement regarding the potential contribution from undetected corrosion and how this compares to the ΔLERF from the requested change (without corrosion).

Response to comment 12, page 7:

A summary statement regarding the potential contribution from undetected corrosion will be added to the report.

13. Section 5.1.5.2, last paragraph: Recommend additional discussion (or entries in Table 5-13) describing the estimated leakage probability values corresponding to the alternative leakage magnitudes of 100 to 600 La.

Response to comment 13, page 7

Additional discussion describing the alternative magnitudes of 100 to 600 La will added to the report.

14. Section 5.1.5.3, 3rd paragraph: The statement "It is likely that an update of the fire analysis would lead to similar changes in total frequency . . ." (as observed in internal events PRA updates) is just speculation. In supporting the assumption that the external events CDF is approximately equal to the internal events CDF, the staff would expect that the analysis compare the original fire CDF with the new internal events CDF without such speculation.

    Response to comment 14, page 7:

    In the absence of quantitative PRAs for other external events (fire, seismic, and other events) such speculation will be required to address external events in any other form than qualitative. Additional justification will be added to the report to support the quantitative aspects of the external event portion of the analysis if this information is available. In the alternative a qualitative analysis will be substituted.

15. Section 5.1.5.3, 4th paragraph: Total LERF is indicated to be equivalent to the sum of the frequency of EPRI Classes 2, 3b, and 8. Per the description of EPRI Classes (Table 4-1), some LERF sequences may also be included in Class 7. Thus, this accounting of LERF is not complete, and should be clarified.

    Response to comment 15, page 7:

    Per Section 4.3, "Class 7 sequences: This group consists of all core damage accident progression bins in which containment failure induced by severe accident phenomena occurs (for example, H_2 combustion and direct containment heating)". It is assumed that class 7 sequences involve significantly degraded core or ex-vessel phenomena. Both of these assumptions result in late timing. While it is true that a small fraction of the cases could before vessel breach, the timing remains relatively late. Not including class 7 accident sequences in the total LERF is conservative in the case of this risk informed application since its inclusion would increase the base case LERF, have no impact on the calculated absolute delta LERF and decrease the delta LERF percent.

16. Section 5.2.1: The frequency of EPRI Class 7a (large, early, unscrubbed) is reported as 5.29E-7, but this value is inconsistent with the frequency of large, early, unscrubbed releases in Table 5-16. An explanation should be provided.

    Response to comment 16, page 7:

    An explanation or correction will be provided.

17. Section 5.2.5.3: In the BWR example application, rather than assuming that all external events could potentially contribute to large leakage (EPRI Class 3b), it was assumed that only the fraction of the external events that would contribute to LERF would be subject to the Class 3b leakage probability. This is non-conservative relative to using the total CDF or the intact containment CDF for external events, and does not represent best practices that should be followed by other licensees applying the EPRI methodology. Further discussion should be provided in the document to address this matter.

    Response to comment 17, page 7:

    Further discussion of this treatment will be provided.

18. Section 6.1: The population dose increase of 11.8 percent in the PWR example is an artifact of the very small conditional containment failure probability for this plant. The report appropriately notes that while this increase is significant on a percentage basis, the total dose remains small. However, this discussion of results should be expanded to include the population dose increase in absolute terms (person-rem per year) and to contrast these values to the population dose increases reported in NUREG-1493.

    Response to comment 18, page 7:

    The discussion will be expanded to include a more complete assessment of the population dose in absolute terms.

19. Section 6.2, 5th paragraph: Only a brief reference is made to "the many analyses developed to date," and the substantial amount of information on these analyses compiled in Appendix G is not effectively used to support the overall conclusions of the EPRI study. Much more could be done here to build a case that, generically, the risk-impact of a permanent, 15-year ILRT test interval would be small.

    Response to comment 19, page 7:

    Additional discussion and more effective use of the data in Appendix G will be added to Section 6.2.

20. Page H-6, next to last paragraph: The document indicates that no criteria have been established for evaluating changes to the population dose parameter. Although a specific value or threshold has not been specified by the staff, the magnitude of a change that can be characterized as "small" can be inferred from both NUREG-1493 and the values cited in previous staff reviews of one-time ILRT extensions.

    Response to comment 20, page 8:

    The report will specify, based on the pervious approved submittals and NUREG-1493, and considering the risk informed application specific ILRT Interval Extension, criteria for small population dose. The report will also indicate the narrow scope of this definition.

21. Page H-6, last paragraph: The methodology and template does not provide sufficient guidance for plants that require containment over-pressure for adequate ECCS net positive suction head. The methodology should indicate that a traditional license amendment request should be submitted for those plants.

    Response to comment 21, page 8:

    The methodology will be revised to emphasize that if containment overpressure is required for ECCS pump operation that the affects of loss of ECCS given a containment leakage event must be reflected in the risk evaluation. In addition, the CLLIP notice will indicate that the CLLIP only applies to those plants that do not credit overpressure for ECCS pump operation and a traditional license amendment require is required in these cases. The template will not change. Please refer to the response for comment No. 10.

22. Page H-7, 1st bullet: The text should be replaced with a statement to the effect that ΔLERF is used to show that the risk acceptance guidelines of RG 1.174 are met, and changes in the population dose and in the conditional containment failure probability are also considered to show that defense-in-depth and the balance of prevention and mitigation is preserved.

    Response to comment 22, page 8:

    Agreed, this is consistent with RG 1.174 R1.

23. Pages H-8 and H-43 (also see comment on Section 4.2.7 of main report): A ground rule should be added to indicate that the risk acceptance guidelines are intended for comparison with a full scope risk assessment, including internal, external, and low power/ shutdown events, and that, consistent with this guidance, the assessment of the impact of the requested change on ΔLERF and total LERF should include consideration of both internal, external, and shutdown events, to the extent supported by the available PRA models. If no such PRA models are available, the licensee should, at a minimum, consider the impact of the requested change on ΔLERF and total LERF (including external and shutdown events) based on a conservative or bounding characterization of the potential contribution from these events.

    Response to comment 23, page 8:

    Agreed. In the new section related to external event treatment in the methodology (see comment response #9) it will be emphasized that the criteria is intended to be applied to a full scope PRA. The guidance will emphasize that if no external event models are available, conservative or bounding analysis or qualitative assessments using judgment will be applied.

24. Page H-12, sentence preceding Section 4.2, and page H-23: All plants will not have a similar containment type. Accordingly, the plant-specific application should address the plant-specific differences from the Calvert Cliffs containment design, and how the methodology for assessing the impact of corrosion was adapted to address the specific design features.

    Response to comment 24, page 8:

    Agree. Recommendations for licensees should also include examination of approved LARs for one time extensions involving similar containment types.

25. Page H-17, population dose calculation: The example calculation in the template should be made consistent with the guidance in Section 4.2.2. For example, in Section 4.2.2 it is stated that the population dose should be adjusted to account for reactor power level and other significant plant-specific features, but this was not done in the example.

    Response to comment 25, page 8:

    Agreed. The template will be modified to account for reactor power level and other significant plant-specific features.

26. Page H-36, table - The annual population dose values reported in this table appear unrealistically high (Indian Point) or lower expected for a typical nuclear power plant. The document should cite more realistic values, such as those that are based on plant and site-specific MACCS2 calculations performed in support of the SAMA analysis for license renewal. These values are typically is the range of tens of person-rem per year.

Response to comment 26, page 8:

Table H-36 provides a summary of the approved submittals to date without modification. It is realized that previous submittals may have been unduly conservative. This will be noted in the table with the additional guidance that the population dose should be based on the most realistic data that can be reasonable obtained.

PRA-A Editorial Comments/Typographical Errors

1. Page 4-1, Section 4.1, 1st paragraph: The sentence fails to identify the fourth area of improvement.

   Response to comment 1, page 9:

   The first sentence will be revised to address the four areas discussed in the paragraphs that follow.

2. Page 4-7, next to last paragraph, 3rd sentence: Change "likely" to "unlikely."

   Response to comment 2, page 9:

   This entire section does not read well. It will be revised to improve clarity.

3. Page 5-9, 1st paragraph, last sentence: Should include reactor power level as another difference that is not accounted for in the preceding calculation.

   Response to comment 3, page 9:

   Agreed.

4. Page 5-11, Table 5-9: The value "2.10E-07" in next to last column should be "2.09E-07."

   Response to comment 4, page 9:

   The difference is round off error. The entry will be corrected.

5. Page 5-14, last bullet: Add the word "large" before "early releases."

   Response to comment 5, page 9:

   Agreed.

6. Page 5-15, Note (4): Delete the word "of" before "probability."

   Response to comment 6, page 9:

   Accepted. This will be changed.

7. Page 5-16, 2nd paragraph, 4th sentence: Move the word "case" to the end of the sentence.

Response to comment 7, page 9:

Accepted. This will be changed.

8. Page 5-17, Table 5-12, row 3b, column 1 per 15 years: The values for frequency without and with corrosion appear inconsistent (it seems that they should be 2.09E-07 and 2.11E-07 respectively, rather than 2.10E-07).

Response to comment 8, page 9:

The table will be reviewed and adjusted as necessary.

9. Page 5-17, Table 5-12, row Class 3b LERF, column 1 per 15 With Corrosion: The "Class 3b LERF" value with corrosion should be "(2.2E-9)" rather than "(1.0E-09)."

Response to comment 9, page 9:

The table will be reviewed and adjusted as necessary.

10. Page 5-22, Table 5-16: The frequency of Containment Failure - Large Early Release (not scrubbed) should be 6.9E-07 rather than 6.7E-07.

Response to comment 10, page 9:

The table will be reviewed and adjusted as necessary.

11. Page 5-22, last paragraph, 2nd sentence: Add the word "this" before "EPRI."

Response to comment 11, page 9:

Accepted. This will be changed.

12. Page 5-24, Class 3a Frequency equation: The value "7.33E-07" should be "7.33E-06."

Response to comment 12, page 9:

Accepted. This will be changed.

13. Page 5-27, next to last paragraph: The last sentence is an incomplete sentence.

Response to comment 13, page 9:

Accepted. "is bin" will be inserted before "#8". ???

14. Page 5-36, Table 5-26, row Class 3b LERF: Several entries are inconsistent with the corresponding values earlier in the table (e.g., "1.23E-08" in the second column should be "1.19E-08," and "5.94E-08" in the sixth column should be "5.95E-08").

Response to comment 14, page 9:

The table will be reviewed and revised as necessary.

15. Page H-1: The third sentence should be broken into two sentences.

Response to comment 15, page 9:

Accepted. Change will be made.

Nuclear Regulatory Research Comments

1. The frequency of Class 3b sequences is taken as a measure of the Large Early Release Frequency (see pages 2-2 and 2-3 of the EPRI report). This is conservative. However, Class 3b corresponds to leak rates greater than or equal to 35 La. As shown in Table 4-1 of the EPRI report, the method of calculating the population dose (per accident) for Class 3b accidents is to assume that Class 3b accidents have exactly a leakage rate of 35 La, not a leakage rate greater than 35 La. Thus, Class 1 is assumed to have a containment leak rate of 1 La, and therefore the population dose (per accident) for Class 3b is assumed to be 35 times the population dose (per accident) for Class 1 accidents. This is not conservative, but leads to an underestimate of the expected population dose. A conservative estimate could be taken by assuming that the containment leakage is that corresponding to a large early release, or 600% per day, as noted on page 3-4 of the EPRI report. Then the population dose per accident for Class 3b accidents would be in the range 600 La to 6000 La, or 600 to 6000 times the population dose for Class 1.

   Please take into consideration the fact that the Class 3b leakage rate exceeds 35 La, and is not equal to it, in your estimate of the change in expected population dose. Supply new risk estimates assuming that the Class 3b leakage rate is 600% per day.

   Response to comment 1, page 10:

   The goal of a risk informed approach is to be as realistic as possible while erring to the conservative when necessary. It can be overly conservative to assess the ILRT failure probability using the Jeffery's non-informative prior and then assess the resulting leakage at a significantly higher level. The Jeffery's non-informative prior is the frequency of exceedance, which in this analysis, has been chosen to represent a leakage magnitude of 35La or greater. The combination of ILRT failure probability (Jefferys Non-Informative prior) and leakage magnitude (35 La) is conservative in that ILRT failures with an La higher than 1.5 have not occurred. The choice of 35 La is based upon an ORNL references for the potential onset of health effects to the public. The method chosen to evaluate the risk impact of extended testing intervals is further complicated by the fact that the figure of merit, LERF, is actually significantly higher than 35 La. While the change is LERF is not a continuous function, population dose is continuous having contributions across the range of La. If we consider just the population from resultant large leakage events that represent LERF, the combination of leakage probability and rates are conservative. However, total population dose is also of interest.

   The optimum way to address the question of ILRT failure probability versus consequence is to develop a relationship between the frequency of exceedance and the magnitude in terms of La. The ILRT failure probability expert elicitation was performed to assess this relationship. In addition, the expert elicitation has other significant advantages over the Jeffery's non-informative prior in that very small containment leakage events have been discovered using the ILRT. The probability of these very small leakage events can be used as a basis for extrapolating the probability of larger leakage events. Another advantage of the expert elicitation is the ability of the process to consider physical aspects in the development of the failure probability such as the phenomena that influence failure, failure modes, alternate inspections and testing, and others.

   In summary, the probability of leakage events is based on the Jeffery's non-informative prior. The leakage rate associated with this probability is assessed at 35 La. Sensitivity cases with

larger leakage rates (greater than or equal to 100 La) will be assessed to address the changes in overall population dose.

2. Table 3-1 of the EPRI report states that 1/n is an upper bound estimate of the failure probability for zero observed occurrences. A classical 95% upper confidence limit is about 3/n. The estimate 1/n corresponds to about a 63% upper confidence limit, which is not really useful as an upper bound. In Table 3-1, a typical range for estimates of failure probability is stated to be from 0.3 to 0.1 for zero failures in n trials. This seems very low. Moreover, the mean of the Jeffreys prior is characterized as a conservative estimate on p. 2-4, Section 2.3 of the EPRI report. It is usually characterized as a best estimate, not a conservative estimate.

   The EPRI report, as well as NEI 94-01 (see Section 11.2) refers to the expert elicitation as indicating that the Jeffreys prior leads to a conservative estimate. The staff, in its meeting with NEI and EPRI on June 17, 2005, noted many concerns about the expert elicitation used in the earlier version of the EPRI report. Without resolving these concerns, the staff cannot accept the expert elicitation results as indicating that the Jeffreys prior leads to a conservative result.

   Justify the characterization of the estimates in Table 3-1 of the EPRI report, or just eliminate the table and state that the mean of the Jeffreys prior is being used, and that the mean of the Jeffreys prior is a best estimate.

   Response to comment 2, page 10:

   While not necessarily in agreement with the comment, the Table 3-1 will be removed and the Jeffery's non-information prior will be indicated as the best estimate.

3. In the first paragraph on page 4-2 of the EPRI report, it is stated that 5/182 is the mean estimate for the probability of failure, given 5 failures out of 182 trials, and moreover that this estimate is more conservative than the 95% upper limit. In classical statistics, 5/182= 0.0274 is the maximum likelihood estimate, not the mean. The 95% upper limit, classically, is 0.057 (Clopper and Pearson upper bound, using the binomial distribution). The maximum likelihood estimate is clearly less than the 95% upper bound. From a Bayesian point of view, the posterior mean, for 5 failures out of 182 trials, and using the Jeffreys non-informative prior for a proportion, is (5+0.5)/(182+1)=0.03, close to the maximum likelihood estimate. [Note that the Jeffreys non-informative prior for a proportion is given by (p-0.5(1-p)-0.5).] The Bayesian 95th percentile of the posterior distribution is 0.05325, clearly larger than the Bayesian mean.

   Justify the statement that the mean estimate exceeds the 95th percentile estimate. Alternately, delete the reference to the mean estimate being greater than the 95^{th} percentile, and characterize the estimate 5/182 as the maximum likelihood estimate.

   Response to comment 3, page 11:

   Reference to the mean estimate being greater than the 95^{th} percentile will be deleted and the 5/182 will be characterized as the maximum likelihood estimate. In addition, additional data collected since the development of the report will be added to Appendix G and a new Jefferys Non-Informative prior will be developed.

4. The absolute change in population dose caused by the lengthening of the ILRT Type A test interval is frequently considered a better measure of the risk increase than is the percent

change in population dose. For example, in cost-benefit analyses, the relevant measure is the (monetized) absolute change in population dose, not the relative change.

Please supply the absolute values of the change in population dose, in addition to the percent change.

Response to comment 4, page 11:

The absolute change in population dose will be added to the methodology.