February 21, 2007

Mr. James H. Riley, Director
Engineering
Nuclear Energy Institute
1776 I Street, NW, Suite 400
Washington, DC 20006-3708

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING NUCLEAR ENERGY INSTITUTE (NEI) 94-01, REVISION 1J, “INDUSTRY GUIDELINE FOR IMPLEMENTING PERFORMANCE-BASED OPTION OF 10 CFR PART 50, APPENDIX J” AND ELECTRIC POWER RESEARCH INSTITUTE (EPRI) REPORT NO. 1009325, REVISION 1, DECEMBER 2005, “RISK IMPACT ASSESSMENT OF EXTENDED INTEGRATED LEAK RATE TESTING INTERVALS” (TAC NOS. MC4235 AND MC9663)

Dear Mr. Riley:

By letter dated December 19, 2005, the NEI submitted for U.S. Nuclear Regulatory Commission (NRC) staff review the NEI Topical Report (TR) 94-01, Revision 1j, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J and Electric Power Research Institute (EPRI) Report No. 1009325, Revision 1, December 2005, “Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals.”

The NRC staff has identified a number of items for which additional information is needed to continue its review. The request for additional information (RAI) questions were discussed with Ms. Julie Keys, Senior Project Manager for NEI on January 9, 2007. It was discussed that the NRC staff would need to receive the responses to the enclosed RAI questions by April 30, 2007. Please call me at 301-415-3610, if you have any questions on this issue.

Sincerely,

/RA/

Tanya M. Mensah, Senior Project Manager
Special Projects Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project Nos. 669 and 689

Enclosure: RAI questions

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

NUCLEAR ENERGY INSTITUTE (NEI)

TOPICAL REPORT (TR) 94-01, REVISION 1J

"INDUSTRY GUIDELINE FOR IMPLEMENTING PERFORMANCE-BASED

OPTION OF 10 CFR PART 50, APPENDIX J"

PROJECT NOS. 689 AND 669

All section, paragraph, page, table, or figure numbers in the questions below refer to items in the NEI TR 94-01, Revision 1j, unless specified otherwise.

Containment and Ventilation Branch

General Comments

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.

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.

Editorial Comments/Typographical Errors

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."

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."

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.

4. Section 10.2.2.1, line 735: The term "P_{ac}" has not been changed to "P_a".

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

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...."

Geosciences and Civil Engineering Branch

General Comments

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.

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.

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.

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.

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.

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. L_a is a

surrogate for an essentially leaktight condition. This type of discussion is appropriate in these lines.

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.

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.

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.

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.

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.

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.

Probabilistic Risk Assessment Licensing Branch A

Please note: In the following comment section the size of the potential containment leakage pathway is expressed as La.

General Comments

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.

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.

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.

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).

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.

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.)

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.

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.

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 licensees 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.

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.

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.

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).

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

Editorial Comments/Typographical Errors

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

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

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.

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

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

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

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

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).

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).”

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.

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

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

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

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").

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

The Office of Nuclear Regulatory Research

Please note: In the following comment section the size of the potential containment leakage pathway is expressed as La.

General 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.

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.

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 95th percentile, and characterize the estimate 5/182 as the maximum likelihood estimate.

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.