February 18, 2004

Mr. Vince Langman
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Atomic Energy of Canada Limited (AECL) Technology, Inc.
481 North Frederick Avenue, Suite 405
Gaithersburg, Maryland 20877

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION - ACR-700 PRE-APPLICATION QUALITY ASSURANCE REVIEW

Dear Mr. Langman:

Atomic Energy of Canada Limited (AECL) submitted a formal request for a pre-application review of the Advanced CANDU Reactor (ACR-700) design on June 19, 2002.

The Nuclear Regulatory Commission (NRC) staff is reviewing technical information provided by AECL as part of the ongoing pre-application review activities for the ACR-700 design. The NRC staff has determined that additional information is necessary to continue the review. The requests for additional information (RAIs) are included in the attachment. The topic covered in these RAIs include the analysis basis of AECL's probabilistic safety assessment methodology of the ACR-700 design. An advanced copy of the RAIs was sent to you via electronic mail on January 20, 2004. On February 6, 2004, AECL participated in a meeting with the staff to discuss the content of the RAIs and agreed to provide the documents containing the ACR-700 information requested in the RAIs by March 31, 2004.

If you have any questions or comments concerning this matter, you may contact the undersigned at (301) 415-4125 or jsk@nrc.gov.

Sincerely,

/RA/

James Kim, Project Manager
New Reactors Section
New, Research and Test Reactors Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 722

Enclosure: As stated

cc: See next page

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ADAMS ACCESSION NUMBER: ML040370701

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Distribution for Request For Additional Information dated February 18, 2004

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REQUEST FOR ADDITIONAL INFORMATION - LETTER 4
ACR-700 PRE-APPLICATION REVIEW - PRA ANALYSIS BASIS

The following questions and comments were generated from an initial review of “Analysis Basis: Probabilistic Safety Assessment Methodology,” AECL Report 108-03660-AB-001, Revision 1, July 2003. The following additional information is required to complete the review.

37. Section 1.1, Page 1-2: This section states that the PSA will satisfy ASME Capability Category I. Based on Table 1.3-1 in ASME RA-S-2002 (which provides the bases for PSA capability categories), Section 2.2.3 of Regulatory Guide 1.174, Section III.2.2.4 in Chapter 19 of the Standard Review Plan (SRP), and Section 1.3 of Regulatory Guide 1.200, the NRC staff believes that most elements of the ACR-700 PSA should meet or exceed Capability Category II. The ACR-700 PSA should identify the relative importance of dominant contributors at the component level, using design-specific data and models to the extent practicable. Any departures from realism should have a small impact on the conclusions and risk insights. The NRC staff notes that ASME RA-S-2002 does not provide a means to determine the overall capability of a PSA; rather, different capability categories are used for various PSA elements. Please provide a self-assessment of the ACR-700 PSA that indicates the expected ASME capability category for each supporting requirement, and provide justification for acceptance of PSA elements that not meet ASME Capability Category II.

38. Section 1.3, Page 1-7: The methodology document states that a Level 1 and Level 2 PSA addressing internal events, internal floods, internal fires, and shutdown states will be performed. However, no technical details have been provided about how specific plant operating states (POS) and off-normal, but periodic changes in plant configurations will be reflected in the PSA. Please explain how POSs and off-normal, but periodic changes in plant configuration will be defined, including those related to shutdown operations and on-power refueling. Describe how the PSA logic model will reflect each unique POS and off-normal, but periodic change in plant configuration.

39. Section 1.3, Page 1-8: Describe the process used to feedback PSA insights to the designers. Is this feedback ongoing, time delayed, or a future activity? In addition, describe how design changes are incorporated into the PSA to ensure that it reflects the actual to-date design. What freeze date is associated with the PSA to be provided as part of the standard design certification application (e.g., three months before the application date)?

40. Section 2.1, Page 2-1 and Section 11.1, Page 11-1: The stated objectives of the PSA imply that the main purpose for developing the PSA is to demonstrate compliance with various numerical risk acceptance guidelines. The NRC staff notes that compliance with numerical risk acceptance guidelines does not mean that the design is acceptable, and that noncompliance does not mean that the design is unacceptable. Therefore, while numerical risk acceptance guidelines are useful tools, the NRC staff believes that the main purpose of performing the PSA is to obtain insights on severe accident vulnerabilities. For example, the PSA should provide information to designers, plant

owners and operators, regulators, and the public about the types of risk-significant accidents and their causes (e.g., human error, common-cause failure, etc.). It is suggested that this section be revised to reflect these uses of the PSA.

41. Section 2.1, Page 2-1: AECL's risk acceptance guidelines do not match those contained in SECY-90-16 (see ADAMS Accession Numbers ML003707849 and ML003707885). AECL has proposed a severe core-damage frequency (SCDF) target of < 1E-5/year, which is an order of magnitude less than the NRC's target for the total core-damage frequency (CDF) of < 1E-4/year. However, AECL has not proposed a target for the limited core-damage frequency (LCDF), although the PSA appears to be capable of estimating the LCDF. In addition, the NRC target for the conditional containment failure probability (CCFP < 0.1) is not discussed in the AECL PSA methodology document.

42. Section 4.2.4, Page 4-2 and Section 4.3.2.1, Page 4-5: Please explain how success criteria are developed from the definitions of limited core damage (LCD) and severe core damage (SCD). Describe the extent of the thermal-hydraulic calculations used to determine success criteria. For example, will all success criteria be based on calculations? If not, what decision process is used to determine that certain calculations are not necessary? Since the NRC has limited experience with CANDU plants, success criteria should have objective bases rather than relying upon engineering judgement that cannot readily be confirmed by the NRC staff.

43. Section 4.2.5, Pages 4-2 to 4-3 and Section 4.4.1, Page 4-9: The methodology document lists three methods for estimating IE frequencies, all of which are statistical in nature. Will fault trees be used to estimate some IE frequencies (e.g., for support systems failures that would initiate a transient)? If so, which ones?

44. Section 4.2.5.4, Page 4-3 and Appendix A, Section A.1: The chi-squared approximation using 2n+1 degrees of freedom is an estimator for the median of the failure rate uncertainty distribution. The PSA should be quantified using the means of the uncertainty distribution.

45. Section 4.3.2.4, Page 4-6 and Section 9.3, Page 9-2:: How does the PSA methodology account for the dynamic effects of high-energy line breaks (pipe whip, jet blast impingement, steam flooding)?

46. Section 4.3.4, Page 4-7: The methodology document states that sequence development is terminated on low frequency. Since sequence development occurs before PSA quantification, how can this approach be practically applied? The NRC staff notes that this approach may work for a baseline risk estimation, but produces a PSA that is inadequate to support future changes to the licensing basis (where some previously terminated sequences may need further development) or real-time risk monitoring. It seems better to completely model all sequences, then let the computer software truncate low frequency sequences.

47. Section 4.3.4, Page 4-7: This section proposes to use a truncation limit of 1E-10/year for accident sequences. However, external events will be screened at 1E-07/year. Please reconcile this difference. Also, will the truncation limit be varied to demonstrate convergence of the accident sequence frequencies?

48. Section 4.3.4, Page 4-7: This section implies that sequences resulting in limited core damage(LCD) will not be addressed in the Level 2 PSA. However, AECL Report 108-126810-LS-001, Chapter 3, Page 3-1 states that LCD accidents will, in fact, be addressed in the Level 2 PSA. The NRC staff believes that the Level 2 PSA should address all core-damage sequences (both SCD and LCD). Please reconcile this difference.

49. Section 4.4.5, Page 4-11: The use of a basic event naming scheme alone will not completely defend against mislabeling errors. The entire set of fault trees must be reviewed, paying particular attention to the system boundaries to ensure that the same basic events in different system models have the same labels.

50. Section 4.4.6.3, Page 4-12: Does this section contain a typographical error ("2 months" is used in the second paragraph whereas "4 months" is used in the third paragraph)?

51. Section 4.4.8, Page 4-14: The systems analysis documentation should contain or reference the basis behind system success criteria.

52. Section 4.5, Page 4-14: This section appears to contain an editorial error since it is entitled "Labeling [sic] of Fault Tree Events," but partially discusses the dependent failure analysis.

53. Section 4.7.2.1.1, Page 4-15: Data will be based "as much as possible" on operating experience at Pickering NGS A and Bruce NPS A. Why only these plants? Why not use data from all CANDU plants?

54. Section 4.7.2.1.2, Page 4-16: This section states that generic data will be obtained from IEEE Standard 500-1984 and NPRDS 1983 Annual Report. Since these data sources are more than 20 years old, how do they apply to an advanced reactor such as ACR-700?

55. Section 4.7.3, Page 4-16: Describe how the "total component operating time" will be obtained. This information should depend on the specific failure mode ("failure to start," "failure to run," etc.).

56. Section 4.7.3, Page 4-17: How will uncertainty in MTTR be estimated and propagated in the logic model?

57. Section 4.8.2.1, Page 4-20: Describe how logic flags (house events) will be defined, incorporated in the logic model, set during accident sequence quantification, and documented.

58. Section 4.8.2.1, Page 4-20: Describe how top logic (fault tree logic used to combine systems for a particular event tree heading) is reviewed and documented.

59. Section 4.8.2.6, Page 4-22: Will fault tree modularization be performed by each system analyst, or done during the PSA quantification effort?

60. Section 4.8.2.8, Page 4-22: Please describe how the intended recovery analysis scheme (applying recoveries only to sequences with frequency > 1E-9/year) avoids skewing the risk profile.

61. Section 4.8.2.8, Page 4-22: Could multiple recoveries be applied to the same cut set? If so, how are potential dependencies among the recoveries addressed? If not, what scheme will be used to prioritize the application of recoveries?

62. Section 4.9.1, Page 4-23: Please clarify the first paragraph of this section. Are all cut sets in a given sequence assigned to the same PDS?

63. Section 4.9.2.7, Page 4-27: AECL should already have adequate information ("The Technology of On-Power Refueling," 108-35000-LS-001, Rev. 0, September 2003, which has 238 pages of information) to analyze PDS10 (fueling machine failures). The NRC staff expects that the PSA will contain adequate modeling to support the quantification of PDS10.

64. Section 5.1, Page 5-2: The most recent NRC guidance on common-cause failure (CCF) methods is contained in NUREG/CR-5485, and the most recent CCF data is NUREG/CR-5497. The cited references should be updated.

65. Section 5.5.1, Page 5-8: The Unified Partial Method (UPM) is a methodology for determining CCF beta factors. It is not capable of modeling partial CCF groups (e.g., two-out-of-three components in a CCF group fail). Ignoring partial CCF groups is not acceptable. Please describe how partial CCF groups will be addressed.

66. Section 5.5.3.2, Pages 5-11 and 5-12: How is uncertainty in CCF event probabilities estimated?

67. Section 5.5.3.1, Page 5-11: Will AECL employ the screening approach to CCF modeling? If so, NUREG/CR-5485, Table 3-1 contains the latest recommendations for beta factors to be used in the screening analysis.

68. Section 5.5.3.2, Table 5-1, Page 5-12: Since human factors are presumed to be addressed in the HRA, they have been removed from the CCF analysis. Has the denominator used to develop the beta factor (the value of 50000) been adjusted/renormalized to remove the human factor contributions? (Otherwise, the beta factor estimates would consistently be too low.)

69. Section 5.5.4, Table 5-2, Page 5-13: Please justify that the list of component types for CCF analysis is adequate. In particular, justify omitting circuit breakers, heat exchangers, strainers, check valves, and relief valves (PORVs, etc.). NUREG/CR-5485 provides CCF data for these components.

70. Section 5.5.5.6, Page 5-17: How will AECL ensure consistency if the subfactor categories in UPM are reassigned? The work is likely to be done by different analysts.

71. General: The NRC staff presumes that the ACR-700 design will utilize state-of-the-art digital control and instrumentation systems. How will software reliability and CCF potential in digital systems be addressed in the PSA?

72. Section 6.3, Page 6-4: What style of emergency operating instructions (EOIs) will be developed for the ACR-700: symptom-based EOIs or event-based EOIs?

73. Section 6.5.5, Page 6-18: For the post-initiator execution errors, what does “maximum time available” mean? Is this the maximum available execution time, or the maximum available time from the compelling signal (including diagnosis time and execution time)?

74. Section 6.5.6, Page 6-19: This section states that completely dependent post-accident human actions will not generally be modeled in the event trees. While this approach may produce acceptable numerical results, it is essential to provide adequate explanation and documentation. It may be better to include completely dependent post accident human actions in the PSA logic model.

75. Section 6.8, Page 6-21: How is the recovery of offsite power addressed in the PSA?

76. Section 6.8.9, Page 6-23: This section states that recoveries will be modeled using the post-accident diagnosis and execution models. Justify using this approach. The NRC staff expects that some recoveries will have higher failure probabilities that estimated using the post-accident HRA models (e.g., recoveries that rely upon knowledge-based behavior).

77. Section 7.1, Page 7-1: SECY-93-087 (see ADAMS Accession Nos. ML003708021 and ML003708056) specifies that bounding analyses be provided for site-specific external events likely to be a challenge to the plant (e.g., river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). However, the ACR-700 PSA methodology focuses only on seismic events, internal fires, and internal floods. What are AECL’s plans for addressing the other types of external events?

78. Section 7.5.1, Page 7-9: How will the high confidence of low probability of failure (HCLPF) values be determined for ACR-specific structures and components (e.g., the calandria)? Since these components are unique to the ACR design, the use of generic fragilities, expert opinion, or screening values needs justification.

79. General: For the seismic, internal fire, and internal flood analyses, AECL should calculate the frequencies of all PDSs, not just those that comprise SCD.

80. Section 8.3, Page 8-3: During the fire analysis, why may it be necessary to use judgement and assumptions to determine cable locations? The NRC staff expects that the actual cable routing has already been designed and will be used in the PSA.

81. Section 8.4.2, Page 8-6: The section states that if a fire could cause several initiating events, AECL will pick the worst one for further analysis. Define “worst,” and justify why multiple initiators caused by a single fire event will not be addressed.

82. Section 9.6.1, Page 9-8: How will maintenance-induced floods be addressed in the PSA? The current AECL approach only considers floods originating from piping failures.

83. Section 10.1, Page 10-1: AECL should perform uncertainty calculations for each plant damage state (PDS), the limited core-damage frequency (LCDF), the severe core damage frequency (SCDF), the large release frequency (LRF), and the conditional containment failure probability (CCFP).

84. Section 10.1.3.4, Page 10-2: The approach to uncertainty analysis only addresses parametric uncertainties. What analyses (e.g., sensitivity analyses) will be performed to address modeling uncertainties?

85. Section 10.2, Page 10-4: AECL should provide importance measures (Fussell-Vesely and risk achievement worth) in the PSA documentation.

86. Section 11.2, Page 11-1: Please define the term “large release” that will be used in the ACR-700 PSA. Note that the NRC has not formally issued such a definition (see the Staff Requirements Memorandum for SECY-93-138, ADAMS Accession No. ML003761015). However, Appendix A to NUREG/CR-6595, Revision 1 (issued August 2003 as a Draft for Comment) provides three working definitions of large early release frequency that could be adapted.

87. Section 11.4, Page 11-3 and Figure 11-1, Page 11-6: The methodology presumes that anticipated transients without scram (ATWS) sequences (PDS0) will have low frequency, so it is not necessary to address them in the Level 2 PSA. The NRC staff disagrees; it is essential to obtain risk insights (both Level 1 and Level 2 PSA) about ATWS sequences.

88. Section 11.6, Pages 11-4 and 11-5: Please clarify what is meant in this section concerning the development of ACR-relevant failure criteria. The text implies that some design details needed to develop the failure criteria (and hence, the source terms) are not readily available. Why not?

89. Section 11.8, Page 11-5: This section states that “The large release frequency will be derived by screening the source terms bins against criteria of Section 11.2.” However, Section 11.2 does not provide any screening criteria.

90. Section 11.8, Page 11-5: This section states that “The large release frequency will be derived by ... identifying the accident with the highest frequency of relevant bins.” Please clarify this statement.

ACR-700

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