August 18, 2005

Mr. Christopher M. Crane
President and Chief Nuclear Officer
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Exelon Generation Company, LLC
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Kennett Square, PA 19348

SUBJECT: LIMERICK GENERATING STATION UNITS 1 AND 2 (LGS) - REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING PROPOSED USE OF ALTERNATE SOURCE TERM (TAC NOS. MC2295 AND MC2296)

Dear Mr. Crane:

By letter dated February 27, 2004, as supplemented by letter dated October 25, 2004, you submitted a request for amendment to the technical specifications for Limerick Generating Station, Units 1 and 2 (Limerick 1 and 2). The amendment would allow for the use of an alternate source term in the LGS design-basis radiological accident analysis.

The Nuclear Regulatory Commission has determined that responses to the RAI enclosed with this letter are necessary in order for the staff to complete its review. The questions in the enclosure are similar to those that were forwarded to you by letter dated June 30, 2005, and discussed with members of your staff in a public meeting on July 14, 2005.

In order to complete our timely review of your amendment request, we request your response within 60 days from the date of this letter. Please note that if you do not respond to this letter within 60 days, we may reject your application for amendment under the provisions of Title 10 of the *Code of Federal Regulations*, Part 2, Section 2.108.

If you have any questions, I can be reached at (301) 415-8474.

Sincerely,

/RA/

Travis L. Tate, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosure: RAI

cc w/encl: See next page

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Travis L. Tate, Project Manager, Section 2
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Docket Nos. 50-352 and 50-353

Enclosure: RAI

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED AMENDMENT REQUEST

FOR IMPLEMENTATION OF ALTERNATIVE SOURCE TERM (AST)

LIMERICK GENERATING STATION, UNITS 1 AND 2 (LGS)

DOCKET NOS. 50-352 AND 50-353

The following questions apply to both units unless otherwise noted. References to attachments of the cover letter refer to your February 27, 2004, application.

1. Please provide all design-basis accident calculations, including all design-basis parameters, assumptions, or methodologies, that were changed in the radiological design-basis accident analyses as a result of the proposed change. If there are many changes, it would be helpful to compare and contrast them in a table. Also, please provide a justification for any changes.

2. Attachment 8 to the application, Table 11b, does not appear to include a leakage pathway currently in the LGS design basis. Per the LGS Updated Final Safety Analysis Report (UFSAR), Section 15.6.5.5.1.2, "Fission Product Transport to the Environment", states that:

   "In accordance with this guidance, and as explained in Section 6.5.3, the LGS evaluation assumes that the mechanisms discussed above will ensure the assumed 50% mixing within the large reactor enclosure at all times during the period when the reactor enclosure pressure is above minus ¼ inch, as well as when it is below. However, it will also be conservatively assumed that there is unfiltered exfiltration at 2500 cfm, in addition to the SGTS exhaust, during periods when the pressure is above minus ¼ inch wg."

   Please include this pathway or provide adequate justification for not including it.

3. Attachment 1, page 42 of 76, Table A, indicates that the LGS analysis conforms to a list of sections within Regulatory Guide (RG) 1.183, including RG 1.183, Section 4.1.2. A review of the proposed changes indicates that strict conformance with the RG for this section does not appear to be correct. The RG states:

   "4.1.2 The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, 'Limits for Intakes of Radionuclides by Workers' (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC [Nuclear Regulatory Commission] staff. The factors in the column headed "effective" yield doses corresponding to the CEDE."

Your application references the report: K. F. Eckerman, et al., “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988 (Reference 20).

The licensee’s proposed new definition is:

“DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same inhalation committed effective dose equivalent (CEDE) as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The inhalation committed effective dose equivalent (CEDE) conversion factors used for this calculation shall be those listed in Table 2.1 of Federal Guidelines Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," ORNL [Oak Ridge National Labatory], 1989, as described in Regulatory Guide 1.183. The factors in the column headed "effective" yield doses corresponding to the CEDE.”

Please explain how the proposed new definition conforms to the RG if the 1989 ORNL report is different from Reference 20 in the RG. Please provide the values used and the justification.

4. Attachment 1, page 57, Section 6.1, states that the activity released through the main stream isolation valves (MSIVs) is the same concentration as that used for evaluating primary to secondary containment leakage. RG 1.183, Section 6.1, states that the leakage should be assumed to be that activity determined to be in the drywell. These assumptions appear to be inconsistent. Previously, when using the Technical Information Document (TID) source term an assumption that the mixing between the drywell and wetwell is instantaneous and not mechanistically modeled may have been found acceptable. Using the AST non-mechanistic modeling is likely not to be found acceptable.

   The staff does not believe the explanation provided in the comments section of page 57 are compatible with the timing assumptions modeled with the AST. Please provide information sufficient to model the time dependent activity used as a source term for the MSIV leakage.

   a. Explain whether the free space in the suppression pool is used to dilute this activity. If so, provide justification for using this volume and also provide the drywell-to-suppression pool free space flow rates versus time and the basis for the flow rate used.

   b. If any drywell-to-wetwell flow is based on the results of thermal-hydraulic analyses performed for the duration of the release, provide a summary of the analyses for staff review, or

   c. Provide justification for this assumption for the duration of the release.

5. Attachment 1, page 25, Section 4.4.3, states that the suppression pool pH is maintained greater than 7. Page 37, Table 15, states that the initial suppression pool pH is 5.3 and that the standby liquid control (SLC) injection is assumed to occur within 13 hours. Please justify how, with an initial pH at 5.3 and SLC initiation at 13 hours, the suppression pool pH is greater than 7 throughout the 30-day accident.

6. Attachment 1, page 11 of 76, gives conflicting information. It states that the LGS post-LOCA (loss-of-coolant accident) direct-shine dose from the Unit 1 14-inch diameter core spray pipe can be managed using administrative controls within the 0.22 rem. Attachment 1, Page 12 of 76 states that "other sources such as reactor enclosure airborne and external cloud and RERS [Reactor Enclosure Recirculation System], SGTS [Standby Gas Treatment System], and CREFAS [Control Room Emergency Fresh Air System] filters are negligible because of shielding, distance or both." Please provide the assumptions, methods, inputs for these analyses, a quantified value for what is considered negligible, and the results of the shielding analyses.

7. In Attachment 6, page 1, Exelon makes a commitment to NUMARC 93-01, Revision 3, Section 11.3.6.5, rather than Section 11.2.6, as specified in technical specification task force (TSTF)-51, Revision 2. Please provide a justification for why Section 11.3.6.5 is a valid substitution for the section stated within the TSTF.

8. Attachment 1, page 12, Section 4.3.1, states that the releases for the radiological consequences analyses are evaluated at full-power conditions. Please confirm that full-power conditions are most limiting or provide justification for why other conditions were not evaluated to determine the most-limiting release conditions.

9. Appendix B to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, establishes quality assurance requirements for the design, construction, and operation of those structure, system, and components (SSCs) that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, “Design Control,” requires that design control measures be provided for verifying or checking the adequacy of a design. Appendix B, Criterion XVI, “Corrective Action,” requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected. Generic Letter (GL) 2003-01, “Control Room Habitability,” addresses current issues with respect to previously assumed values of unfiltered inleakage. Generally, these issues can only be resolved by inleakage testing.

   Exelon requested a change in the design basis of the control room heating, ventilation, and air conditioning (HVAC) system. This request no longer takes credit for the automatic initiation of the radiation isolation mode. With no credit for this initiation, during the initial 30 minutes of the accident, the control room HVAC operates in the normal mode rather than in the radiation isolation mode. The licensee assumed that in this mode 525 cfm of unfiltered inleakage in addition to the normal 2100 cfm of unfiltered inleakage is transferred into the control room. According to the LGS response to GL 2003-01, this mode of operation does not appear to have been tested for inleakage. In light of the Appendix B requirements and GL 2003-01, provide justification to explain why the value assumed for the control room’s unfiltered inleakage is appropriate. Please provide details regarding control room, design, maintenance and assessments, or tests to justify the use of this number. Please note that because of the high percentage of control rooms that have historically been unable to successfully predict the amount of unfiltered inleakage, the staff will generally only accept a measured value.

10. Starting on page 17 of Attachment 1 of the application, Exelon describes the methodology used to calculate the leakage from the primary containment into the main steam lines. At upstream conditions, the flow rate out of the MSIVs is adjusted by the MSIV surveillance pressures. This method does not appear to consider the accident conditions in the drywell. Methods acceptable for calculating the accident pressures in the drywell typically use the design pressure for this calculation. Please justify the methodology proposed.

11. Page 18 of Attachment 1 states that the inboard steamline, outboard steamline and condenser effective filter efficiencies are calculated using AEB 98-03 formulations and settling and deposition velocities. The discussion and the data provided are insufficient to support an NRC staff confirmation. Please provide the following information.

    a. On page 19 of Attachment 1 of your submittal, you state that your submittal is based upon the methodology used in AEB 98-03.

       If the analysis did not use the entire methodology, please describe differences between the model used and the AEB 98-03 model. Please provide justification for any differences between the two models.

    b. A single-line sketch of the four main steamlines and the isolation valves. Annotate this sketch to identify each of the control volumes assumed by Exelon in the deposition model.

    c. A tabulation of all of the parameters input into the proposed AEB 98-03 model for each control volume shown in the sketch (and time step) for which you are crediting deposition. This includes:

       - Flow rate
       - Gas pressure
       - Gas temperature
       - Volume
       - Inner surface area
       - Total pipe bend angle

       Note: Attachment 8, Table 4, provides some of this information but neither the paper nor the electronic copy of this file is legible.

    d. For each of the bulleted parameters in this question, please provide a brief derivation and an explanation of why that assumption is conservative for a design basis calculation. Address changes in parameters over time, e.g., plant cooldown.

e. Page 46 of Attachment 1, Table A, indicates that the Exelon analysis conforms to RG 1.183, Section 5.1.2. Please clarify whether your analysis addresses a single failure of one of the MSIVs. Such a failure could change the control volume parameters that are input in the deposition model. Previous implementations of main steam deposition have been found acceptable only if the licensee modeled a limiting single failure. Please confirm that the limiting MSIV single failure has been modeled and describe which failure was utilized and justify why this is the limiting failure.

f. Since crediting the main steamline deposition effectively establishes the main steam piping as a fission product mitigation system, the staff expects the piping to meet the requirements of an engineered safety feature (ESF) system, including seismic and single failure considerations. Please confirm that the main steam piping, condenser and the isolation valves that establish the control volumes for the modeling of deposition were designed and constructed to maintain integrity in the event of the safe shutdown basis earthquake for LGS. If the design basis for the piping and components does not include integrity during earthquakes, please provide an explanation of how the LGS design satisfies the prerequisites of the staff-approved NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems."

If piping systems and components at LGS were previously found by the staff to be seismically qualified using the methodology of this Boiling-Water Reactor Owners Group (BWROG) report, please provide a specific reference to the staff's approval.

g. Page 19 of Attachment 1 states, "For aerosol settling, only horizontal piping runs are credited, and only the bottom surface area is considered available." If only horizontal piping runs are credited, please justify using the surface area of the bottom half of the pipe for aerosol deposition when the cross-sectional edges of this piping are essentially vertical or inclined.

h. Page 10 of Attachment 8 states, "For the two bounding steamlines modeled, two nodes are used." Please specify which two steamlines are bounding and specify how they were chosen and why they are bounding.

i. Table B contained on page 53 of Attachment 1 states that a previous analysis based upon Technical Information Document (TID)-14844-based source terms assumed a recirculation line break. The design loss-of-coolant accident (LOCA) analyses are required by regulation to consider a spectrum of break locations and break sizes. Proposals to credit deposition in the main steamlines need to consider the impact of the break location on steamline deposition. In light of crediting this deposition, please justify why a break of a main steamline is not considered and why the recirculation line remains bounding or consider the break in the most-limiting reactor coolant system location. Note that, although thermodynamic analyses may show that significant core damage is unlikely for a reactor coolant system break in the steamline, a LOCA involving a recirculation piping break is similarly unlikely to cause significant core damage. Nonetheless, the regulatory guidance for a design-basis LOCA assumes a substantial release of fission products as a means of assessing the ability of the containment design to mitigate the consequences of a LOCA in the unlikely event the emergency core cooling system (ECCS) should fail. As such, the break location and size are not determinants for the amount of fuel damage assumed to occur in the stylized design-basis analysis.

j. Page 20 of Attachment 1 states, "Iodine resuspension from settled or deposited iodines is not calculated. Historically, this phenomena increased the organic iodine release by about a factor of two based on resuspension of TID-14844-based elemental iodine fractions. The presence of this phenomenon is questionable with aerosols with significant cesium loadings. Furthermore, while deposition on condenser tubing is not formally credited, test cases have shown that substantial removal of elemental and even organic iodine would be predicted that would more than offset any resuspension. Flow rates out of the condenser are assumed to be at 120 degrees F and atmospheric pressure. A factor of 1.25 is applied, as is done with leakage and flow-through steamlines. This leak rate is also reduced by 50% after 24 hours, consistent with the change in containment conditions."

The staff believes that the above information does not provide adequate justification for changing the historical basis for organic iodine resuspension. Please provide additional information to justify not utilizing the historical resuspension, including the mechanics for changing the current methodology. As a minimum, the information previously used to determine the factor of two should be examined, and LGS should provide a complete assessment of why the previous assessment is no longer applicable.

If reliance on the condenser tubing is being used to offset the change in methodology, then provide a justification that this is conservative. Please confirm that the condenser piping credited is designed and constructed to maintain integrity in the event of the safe shutdown basis earthquake for LGS. If the design basis for the piping and components does not include integrity during earthquakes, please provide an explanation of how the LGS design satisfies the prerequisites of the staff-approved NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems." If the condenser piping systems and components at LGS were previously found by the NRC staff to be seismically qualified using the methodology of this BWROG report, please provide a specific reference to the staff's approval.

12. Page 46 of Attachment 1, Table A, states that the Exelon analysis conforms to Regulatory Position 5.1.2. For each design-basis accident analyzed please provide:

a. The single active component failure that results in the most-limiting radiological consequences and justify why it is the most-limiting.

b. The assumptions regarding the occurrence and timing of a loss of offsite power (LOOP) and justify why it provides the most limiting radiological consequences.

13. Based on information provided in the application, the licensee has assumed an MSIV leakage rate of 0.668 cfm for the 100 scfh lines (0.834 cfm with 25% design-margin included). The leakage rate is reduced after 24 and 96 hours based upon changing steamline temperatures. When the proposed MSIV leakage, in scfh, at test conditions (typically 70 degrees and 25 psig) are scaled to peak drywell pressure and temperature (typically 40-50 psig and about 340 degrees) the TS leakage past the inboard MSIV has been shown to be 2.0 cfm, about double the value assumed. However, the temperature of the fluid in the steamlines is based on the steam piping temperatures, typically 500-600 degrees (558 degrees F for 0-24 hours for LGS). At the steam piping conditions, the flow is even higher. Likewise a pressure gradient will exist from the first closed MSIV to the end of the last deposition. The gradient would depend on the actual leakage through each MSIV. As such, the deposition nodes downstream of the first MSIV conservatively may be assumed to be at atmospheric pressure. Therefore, these flow rates would be even higher. While the trend of increasing flowrates is reflected in Table 4 of the submittal (Attachment 8, page 12), the absolute values calculated by the licensee are smaller than expected when compensating for the changes in temperature and pressure. The equation provided in Attachment 8, page 9, does not adequately compensate for the leakages in the steamline nodes. Likewise, the arbitrary 25% design-margin added, while conservative, does not compensate for the expected flow rates.

    a. Please provide the methodology used to calculate the flow rates in each steamline node and the parameters used. Justify how these parameters conservatively model the changing conditions in the steamline or provide calculations that conservatively account for these steamline-condition changes.

    b. Attachment 1, page 18, states, "However, to provide design-margin, the above leak rate is increased by 25% for the first 24 hours to a value of 0.834 cfm. This margin also allows MSIV leakage to be reduced by 50% at 24 hours." Please explain how the design-margin allows the MSIV leakage to be reduced by 50% at 24 hours.

    c. Page 11 of Attachment 8 provides a generic assessment of the steamline temperatures following a LOCA. Please provide justification for why this generic assessment is applicable and conservative for LGS. Provide References 28 and 29 from the amendment request.

14. In Attachment 1, page 24, a value for the emergency core coding system (ECCS) flash fraction is given as 1.39% as opposed to 10% in RG 1.183, Section 5.5. LGS states that a smaller amount (than the RG) was determined using a method approved for the Clinton Power Station, Unit I. If this value is not in your current licensing basis, please explain why this method is acceptable for LGS. If the value is new, please provide the details used to calculate this value, including the following:

a. Although the analysis includes a limiting pH, no specific details regarding the pH history versus time are provided. Please provide the iodine concentration in the sump versus time. Please provide the pH vs. time or the pH assumed for the duration of the accident, including justification for the pH and iodine concentration used. Please provide the area ventilation rates that the ECCS leakages are exposed to.

b. The ORNL study cited in the Clinton AST submittal is based upon theoretical calculations for the design of reactor containment spray systems. Many of the release mechanisms and other plant-specific issues have not been addressed. This creates notable uncertainties in how much iodine is available for release. Major uncertainties exist to what extent the chemicals within the leakage will interact when their release to the environment leads to a great reduction in vapor pressure.

   The production of elemental iodine is related to the pH of the water pools. A major uncertainty in fixing the production of volatile iodine chemical forms is due to uncertainty in the extent of evaporation to dryness. Experts believe that up to 20% of the iodine in water pools that has evaporated would be converted to a volatile form (most likely as elemental iodine). Uncertainties also depend upon the environment where the fluid is leaked and the way the fluid is leaked (misting etc.). Fluid pH shifts may occur due to interactions with components, cable jackets, concrete and radiation. Please include a discussion of these issues to support the proposed value.

15. From the LGS UFSAR, Table 6.2-4a, (stated as Rev. 11), the minimum suppression pool free airspace is given as 147,670 cubic feet. Typically, a Mark II suppression pool free volume is on the order of hundreds of thousands of cubic feet. Please clarify whether the UFSAR number is a typographical error and whether the decimal should be a comma.

    Please provide justification for the use of 159,540 cubic feet provided in Table 3, on page 31 of Attachment 1. Why is the more conservative UFSAR value not valid for the LOCA analysis?

16. Page 61 of Attachment 1, Table C, contains a comparison of the LGS analysis to Section 2.0 of RG 1.183. The LGS analysis column of this table states that it conforms with RG 1.183, but this RG does not find the use of a Decontamination Factor (DF) of 200 acceptable for less than 23 feet of water covering a damaged fuel assembly.

    a. Please provide the DF used for 21.6 and 22.6 feet of water and the parameters, methodology and justification used to calculate this value.

    b. Considering the statement "The conservatively determined damage over the spent fuel pool is 70% of the reactor vessel," please provide the analysis used to justify this statement.

c. The UFSAR fuel handling analysis states that 212 are assumed damaged as the result of the fuel-handling accident. Attachment 1, page 60, states that based upon a generic evaluation of GE11 and GE14 fuel, such an accident "yields 172 failed rods." Is this a change? If so, please justify. If not, where is it substantiated?

17. Attachment 1, page 45, Section 4.2.3, states that the models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel. It states that the LGS analysis conforms to this guidance.

a. Attachment 8, page 7, states that RADTRAD was used to determine the core spray line dose rates. The 60 radionuclides that are contained in the RADTRAD code were selected based upon a study that determined that those 60 radionuclides have the greatest impact on offsite dose. Please confirm that the most conservative radionuclides were used to determine the source for the LGS shielding studies for the shine doses from external sources to the control room. Provide the source terms used and the geometry and materials used in these shielding studies.

b. Attachment 8, page 6, states that a zone is identified where controls are practical and suggests that the maximum boundary dose (at the inside control room wall) from outside sources should not be used to determine the limiting control room dose. Administrative controls and occupancy factors within zones seem to be credited. The value added to the control room dose from gamma shine is .22 rem, which appears to correlate to a dose 18 feet from the wall.

The above described methods and assumptions are inconsistent with your current licensing bases. UFSAR, Section 6.4.2.5, states that shielding is designed for continuous occupancy. Section 12.3.2.3 states:

"The shielding thicknesses are selected to reduce the aggregate radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements are evaluated at the point of maximum radiation dose through any wall. Therefore, the actual anticipated radiation levels in the greater region of each plant area are below this maximum dose and, therefore, below the radiation zone upper limit."

The NRC staff does not find the proposed practice acceptable. Access is needed to these locations. Administrative controls within the control room boundary are not an adequate substitute for potentially inadequate shielding. The staff believes that this is not consistent with the licensee's stated conformance with Regulatory Position 4.2.3 of RG 1.183. Please include the maximum doses from these external sources consistent with your current licensing basis or provide additional justification why such deviations from standard shielding practices are unavoidable and necessary.

c. The licensee states that MicroShield was used to determine the doses from the external piping. Please explain how the impact of scattering is considered. Please justify why modeling of scattering off piping, air, etc., is not considered, or include the impact of scattering in your assessment.

d. Please provide a copy of the calculation or the information necessary to model the shine from this pipe. Include the geometry (drawings, piping, etc.), source term, materials, and assumptions used to determine the doses given on page 7 of Attachment 8.

e. UFSAR, Section 6.4.4.1, states, "Control room shielding design, based on the most limiting radiological accident (design basis LOCA) is discussed in Section 12.3. The evaluations in Chapter 12 demonstrate that radiation exposures to control room personnel originate from containment shine, external cloud shine, and containment airborne radioactivity sources. Total exposures resulting from the worst radiological accident are below the dose limits specified by [General Design Criterion] GDC 19; the portion contributed by containment shine and external cloud shine is reduced to a small fraction of the walls which surround the control room."

   Page 6 of Attachment 8 to the application states that historically the dose due to the core spray piping and other lesser piping contributors is 4.2 rem whole body. The licensee also states that the "Other sources such as reactor enclosure airborne and external cloud and RERS, SGTS, and CREFAS filters are negligible because of shielding, distance or both."

   Please clarify whether the proposed change involves a change to the bases for current shine analysis for piping and sources other than the containment spray piping. If parameters or assumptions have changed, please provide the bases for the sources used, the parameters used for this reevaluation, any assumptions used, and the results of the analyses.

18. Table 11c, page 24, of Attachment 8 to the application indicates that "pathway 6" provides a flow path from node 5 to node 3. Table 11a does not provide a description of node 5. Please provide a description of node 5 (as is done with nodes 1 through 4) and describe how this is different from node 2. If node 5 is the same as the node 5 described in Table 13a of Attachment 8, justify the use of a node for the SGTS. Typically, the SGTS is modeled as a transfer pathway rather than a node. Confirm this model yields conservative results.

19. More detail regarding the main steamline break (MSLB), fuel handling, and control rod drop accidents is needed. Please provide all assumptions, inputs, models and methodologies (CRDAs) used to calculate the offsite and control room doses. Please include answers to the following questions:

    What is the reactor coolant system (RCS) activity used for the MSLB analysis? Provide the assumptions, input, and methods used to determine this activity.

The second bulleted item on page 20 of Attachment 1 to the application states that the activity in the steam cloud is based on the total mass of water released from the break. Confirm that the total activity released for this accident is the RCS-specific activity times the break discharge mass (103, 785 lbm). If this is not the methodology used, please provide more detail regarding the model utilized. Also, provide the input parameters used to calculate and justify the fraction of liquid water contained in the steam and the flashing fraction of liquid water released.

20. In Attachment 1, page 35, Table 8, a value of 0.77% damaged fuel with melt is provided for the CRDA. The value typically used for fuel melt with General Electric 14 fuel is 1% for the CRDA. Please confirm this value of 0.77% and justify the value if this is a change to your licensing basis.

21. Attachment 1, page 16, states that, "Infiltration following isolation is assumed to be 525 cfm of unfiltered inleakage, which includes impacts of ingress and egress." Please confirm that the 525 cfm value includes 10 cfm for the ingress and egress into the control room after a LOCA.

22. Comments provided for Section 5.1.3 in Attachment 1, page 47, state that, "conservative assumptions are used."

    Please confirm that the control room and SGTS HVAC flow rates assumed in the accident analysis (including control room doses) are conservative based on the range of flow rates allowable by the TSs.

23. The proposed change to TS 3.6.5.1.2, “Refueling Area Secondary Containment Integrity,” will no longer require that the secondary containment be operable during the movement of fuel assemblies that have a decay period of at least 24 hours. The fuel-handling accident (FHA) analysis assumes the release to the control room intake and the environment is through the turbine building/reactor building (TB/RB) ventilation south stack. Please justify that an FHA release through the TB/RB ventilation south stack is an appropriately conservative assumption given that the secondary containment may be inoperable. Include general arrangement drawings in your response showing the potential release points.

24. Please explain in detail the methodology used to model steam cloud transport for the MSLB accident. Please also describe the methodology (e.g., inputs and assumptions) used to determine the control room doses for the MSLB accident.

25. The inleakage of unfiltered air into the control room (which can occur through the control room boundary, system components, and backflow at the control room doors) was modeled using the control room intake χ/Q values. Please verify that there are no potential unfiltered inleakage pathways during the normal operation mode, radiation isolation mode, and chlorine isolation mode that could result in χ/Q values that are higher than the control room intake χ/Q values.

26. Provide a curve of containment pressure as a function of time for the large break LOCA to verify that the containment pressure decreases to less than 50% of its peak value within 24 hours.

27. In TS Table 3.3.2-1, “Isolation Actuation Instrumentation Action Statements,” and TS Table 4.3.2.1-1, “Isolation Actuation Instrumentation Surveillance Requirements,” does the instrumentation referenced in the proposed change provide protection for an area that is common to both units and as such would it still be required when either unit was operating even though the other unit is in refueling? Please explain whether the alarm capability of this instrument would be available even if the actuation function were removed? Please explain whether the removal of this function would support the monitoring requirements of GDC-64? Please explain whether procedures are available that would manually isolate in lieu of the automatic isolation that is to be removed?

28. Please explain whether the instrumentation in TS Table 3.3.7.1-1, “Radiation Monitoring Instrumentation,” and TS Table 4.3.7.1-1, “Radiation Monitoring Instrumentation Surveillance Requirements,” would be operable if either unit were operating during an accident in one unit since the control room is common to both units? Explain whether the alarm and isolation functions would still be required since an accident at the operating unit could affect the habitability of the main control room?

29. Considering TS Section 3.6.5.2.2, “Refueling Area Secondary Containment Automatic Isolation Valves.” TSTF-51 allows certain engineered safety feature (ESF) functions to be inoperable, such as the automatic isolation feature; however, it still requires the ability to isolate the secondary containment in order to meet the objectives of NUMARC 93-01. Will the ability to isolate the containment be retained if the automatic feature is disabled? If the secondary containment cannot be isolated, please explain how the station will meet the intent of GDC-64 in monitoring releases and the GDC 61 intent of controlling releases through containment, confinement, or filtering.

30. In TS Section 4.6.5.3, “Standby Gas Treatment System - Common System,” the staff notes that the TS cited references RG 1.52, Revision 2. Revision 2 states the maximum penetration for a 2-inch carbon adsorber should be less than 1%. The staff has issued Revision 3 which allows a penetration of 2.5% for a 2-inch bed filter. Please provide the appropriate RG Revision proposed for the penetration and include any extenuating circumstances where the conditions of the RG are not being met.

    Discuss whether the filter is larger than a 2-inch bed filter. Discuss any specific need to retain RG 1.52, Revision 2, and exceed the maximum penetration limits shown in Table 2.

31. In SR 4.6.5.4.a, explain how this reactor enclosure recirculation system flow rate compares to the design flow rate of the system used in the evaluation of design-basis accidents. Are there any reasons why the design flow rate (rated flow) of the system with an appropriate tolerance should not be specified? Explain whether the proposed change would allow testing at a flow rate that was significantly lower than the design flow rate for its intended service.

32. Please explain whether the changes proposed in SR 4.6.5.4.b.1, SR 4.6.5.4.d.1, SR 4.6.5.4.e, and SR 4.6.5.4.f, provide for doing anything different from the way it is done now.

33. In SR 4.6.5.4.b.2 and SR 4.6.5.4.c, the existing penetration of 2.5% is the maximum allowable penetration for a 2-inch filter based on the conditions of RG 1.52, Revision 3. The TS references Revision 2. The proposed 15% penetration indicates that the carbon adsorber is in a degraded state. The RG 1.52 values are based on clean carbon adsorbers. The staff does not have data to show how quickly carbon adsorbers degrade once they are in a degraded state. Although the analysis may show that a 15% penetration would be acceptable, there is an increased uncertainty that the filters would still be acceptable at the end of the inspection interval. The fact that the filters have reached the degraded state may indicate that some operational changes need to be made to prevent filter degradation. Please provide data to justify the filter performance from a 15%-degraded state for the entire inspection interval or justify this change by other information.

34. In SR 4.6.5.4.b.3, the subsystem flow rate affects the clean-up rate for filtration and should be established at the flow rate credited for the subsystem in any analyses. Please clarify why a large range is needed and why the flow rate cannot be closely tied to the values used in the design-basis analyses.

35. In TS Section 3.7.1.2, “Emergency Service Water System - Common System,” and TS Section 3.7.1.3, “Ultimate Heat Sink,” the staff is concerned that the proposed change does not “expand the definition” as stated. The relaxations that have been granted through TSTF-51 were based on satisfying the requirements of the FHA. Please provide additional justification for this change. Discuss whether other potential transients that would require the use of either the emergency service water or ultimate heat sink have been evaluated to assure that eliminating this applicability is justified with respect to two unit operability in which one unit is at full power.

36. In TS Section 4.7.2, “Control Room Emergency Fresh Air Supply System - Common System,” the existing penetration of 2.5% is the maximum allowable penetration for a 2- inch filter based on the conditions of RG 1.52, Revision 3. The TS references Revision 2. The proposed 10% penetration indicates that the carbon adsorber is in a degraded state. RG 1.52 values are based on clean carbon adsorbers. The staff does not have data to show how quickly carbon adsorbers degrade once they are in a degraded state. Although the analysis may show that a 10% penetration would be acceptable, there is an increased uncertainty that the filters would still be acceptable at the end of the inspection interval. The fact that the filters have reached the degraded state may indicate that some operational changes need to be made to prevent filter degradation. Please provide data to justify the filter performance from a 10%-degraded state for the entire inspection interval or justify this change by other information. Please provide additional justification for changing to a manual initiation of the radiation mode of the control room emergency fresh air system. RG 1.183 states that “modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions.” Please discuss the impact of this change on one unit when the other unit is at full power.

37. Please provide a description of the analysis assumptions, inputs, methods, and results that show that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 within 24 hours of the start of the event. (See also Position 2 of Appendix A to RG 1.183.) In your response, please discuss the adequacy of recirculation of suppression pool liquid via ECCS through the reactor vessel and the break location and back to the suppression pool in meeting the transport and mixing assumptions in the chemical analyses. Assume a large break LOCA.

    In responding to this question, please indicate the source and volume flow rate of water that mixes with the sodium pentaborate and washes it from the vessel to the suppression pool. A diagram showing the injection point of the sodium pentaborate, the flow path through the core, and the exit path from the vessel would be helpful. Please discuss how the proposed change would continue to ensure that the core ECCS flow does not bypass the region of the vessel that contains sodium pentaborate and that sufficient sodium pentaborate will be transported to the suppression pool.

38. The submittal states that LGS is committing to NUMARC 93-01 which requires prompt closure of containment and control of releases from FHAs. NUMARC 93-01 states, in part, that, "these prompt methods need not completely block the penetrations nor be capable of resisting pressure, but are to enable the ventilation systems to draw from the postulated FHA such that it can be treated and monitored." Please describe the prompt methods and the degree of closure that will be achieved. How much of an open area to the environment would be permitted? Also, please describe the ventilation systems that would be used to draw the release from the postulated FHA. Specifically, are the ventilation systems ESF systems, do they have carbon adsorber filters and high-efficiency particulate air (HEPA) filters, are they tested in accordance with RG 1.52 or other standards, and do they have sufficient drawing capacity to assure that air flow is going from the environment to the containment?

    Will there be a test to determine that all air flow was going into the containment in the event that the LGS procedure allows partial closure?

39. Limerick has proposed to credit control of the pH in the suppression pool following a LOCA by means of injecting sodium pentaborate into the reactor core with the SLC system. The SLC system design was not previously reviewed for this safety function (pH control post-LOCA). Please demonstrate that the SLC system is capable of performing the pH control safety function assumed in the AST LOCA dose analysis.

    The following questions are from a set of generic questions developed by the staff and are being provided to all BWR licensees with pending AST license amendment requests. In responding to questions regarding the SLC system, please focus on the proposed pH control safety function. The reactivity control safety function is not in question. For example, the SLC system may be redundant with regard to the reactivity control safety function, but lack redundancy for the proposed pH control safety function. If you believe that the information was previously submitted to support the license amendment request to implement AST, you may refer to where that information may be found in the documentation.

40. Please state whether or not the SLC system is classified as a safety-related system as defined in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.2, and whether or not the system satisfies the regulatory requirements for such systems. If the SLC system is not classified as safety-related, please provide the information requested in Items 1.1 to 1.5 below to show that the SLC system is comparable to a system classified as safety-related. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control agent injection.

    a. Is the SLC system provided with standby AC power supplemented by the emergency diesel generators?

    b. Is the SLC system seismically qualified in accordance with RG 1.29 and Appendix A to 10 CFR Part 100 (or equivalent used for original licensing)?

    c. Is the SLC system incorporated into the plant's American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, inservice inspection and inservice testing programs based upon the plant's code of record (10 CFR Part 50.55a)?

    d. Is the SLC system incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65?

    e. Does the SLC system meet 10 CFR 50.49 and Appendix A to 10 CFR Part 50 (GDC-4, or equivalent used for original licensing)?

41. Please describe proposed changes to plant procedures that implement SLC sodium pentaborate injection as a pH control additive. In addition, please address Items 2.1 to 2.5 below in your response. If any item is answered in the negative, please explain why the SLC system should be found acceptable for pH control additive injection.

    a. Are the SLC injection steps part of a safety-related plant procedure?

    b. Are the entry conditions for the SLC injection procedure steps symptoms of imminent or actual core damage?

    c. Does the instrumentation cited in the procedure entry conditions meet the quality requirements for a Type E variable as defined in RG 1.97, Tables 1 and 2?

    d. Have plant personnel received initial and periodic refresher training in the SLC injection procedure?

    e. Have other plant procedures (e.g., emergency response guidelines/senior advisory groups (ERGs/SAGs)) that call for termination of SLC as a reactivity control measure been appropriately revised to prevent blocking of SLC injection as a pH control measure. (For example, the override before Step RC/Q-1, *"If while executing the following steps: ....it has been determined that the reactor will remain shutdown under all conditions without boron, terminate boron injection and...."*)

42. Please provide a description of the analysis assumptions, inputs, methods, and results that show that a sufficient quantity of sodium pentaborate can be injected to raise and maintain the suppression pool greater than pH 7 within 24 hours of the start of the event. (See also Position 2 of Appendix A to RG 1.183.) In your response, please discuss the adequacy of recirculation of suppression pool liquid via ECCS through the reactor vessel and the break location and back to the suppression pool in meeting the transport and mixing assumptions in the chemical analyses. Assume a large break LOCA.

43. Please show that the SLC system has suitable redundancy in components and features to assure that for onsite or offsite electric power operation its safety function of injecting sodium pentaborate for the purpose of suppression pool pH control can be accomplished assuming a single failure. For this purpose, the check value is considered an active device since the check valve must open to inject sodium pentaborate. If the SLC system cannot be considered redundant with respect to its active components, the licensee should implement one of the three options described below, providing the information specified for that option for staff review.

    a. Option 1 Show acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components. If you choose this option, please provide the following information to justify the lack of redundancy of active components in the SLC system:

        a.1 Identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number.

        a.2 Provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields.

        a.3 Indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, provide information on the quality standards under which it was purchased.

        a.4 Provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS.

        a.5 Provide a description of the component's inspection and testing program, including standards, frequency, and acceptance criteria.

        a.6 Indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. In your response, please consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate when non-redundant active components fail to perform their intended functions.

b. Option 2 Provide for an alternative success path for injecting chemicals into the suppression pool. If you chose this option, please provide the following information:

b.1 Provide a description of the alternative injection path, its capabilities for performing the pH control function, and its quality characteristics.

b.2 Do the components which make up the alternative path meet the same quality characteristics required of the SLC system as described in Items 1.1 to 1.5, 2, and 3 above?

b.3 Does the alternate injection path require actions to be taken in areas outside the control room? How accessible will these areas be? What additional personnel would be required?

c. Option 3 Show that 10 CFR 50.67 dose criteria are met even if pH is not controlled. If you choose this option, demonstrate through analyses that the projected accident doses will continue to meet the criteria of 10 CFR 50.67 assuming that the suppression pool pH is not controlled. The dissolution of gesium iodide (CsI) and its re-evolution from the suppression pool as elemental iodine must be evaluated by a suitably conservative methodology. The analysis of iodine speciation should be provided for staff review. The analysis documentation should include a detailed description and justification of the analysis assumptions, inputs, methods, and results. The resulting iodine speciation should be incorporated into the dose analyses. The calculation may take credit for the mitigating capabilities of other equipment, for example the SGTS, if such equipment would be available. A description of the dose analysis assumptions, inputs, methods, and results should be provided. Licensees proposing this approach should recognize that this option will incur longer staff review times and will likely involve fee-billable support from national laboratories.

44. Page 16 of Attachment 1 states "the transfer of radioactive gases into the control room are minimized by maintaining the control room at a positive pressure of 0.1-inch water column with respect to adjacent areas during emergency pressurized modes." Unit 1 TS 3.7.2 states that the control room is maintained at 1/8-inch water gauge positive pressure. This is equivalent to 0.125-inch water column. Verify that the 0.1-inch water gauge was inadvertently truncated and that LGS is not requesting to change its license basis to 0.1-inch water gauge or provide justification for your proposed change.