April 27, 2006

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

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

Dear Mr. Crane:

By letter dated February 27, 2004, as supplemented by letters dated October 25, 2004, and October 10, 2005, Exelon Generation Company, LLC submitted a request for an amendment to the Technical Specifications for Limerick Generating Station, Unit Nos. 1 and 2 (LGS). The amendment would allow for the use of an alternative source term in the LGS design-basis radiological accident analysis.

The Nuclear Regulatory Commission has determined that responses to the enclosed Request for Additional Information are necessary in order for the staff to complete its review. The questions in the enclosure are similar to topics that were discussed with members of your staff during a public meeting on January 30, 2006, and in subsequent teleconferences.

In order to complete our timely review of your amendment request, we request your response within 30 days from the date of this letter.

If you have any questions, please contact Theresa Valentine at 301-415-4048.

Sincerely,

/RA by Theresa Valentine for/

Richard V. Guzman, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosure:
Request for Additional Information

cc w/encl: See next page

Limerick Generating Station, Unit Nos. 1 and 2

cc:

Site Vice President
Limerick Generating Station
Exelon Generation Company, LLC
P.O. Box 2300
Sanatoga, PA 19464

Plant Manager
Limerick Generating Station
Exelon Generation Company, LLC
P.O. Box 2300
Sanatoga, PA 19464

Regulatory Assurance Manager - Limerick
Exelon Generation Company, LLC
P.O. Box 2300
Sanatoga, PA 19464

Senior Vice President - Nuclear Services
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Vice President - Operations, Mid-Atlantic
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-N
Kennett Square, PA 19348

Vice President
Licensing and Regulatory Affairs
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

Director
Licensing and Regulatory Affairs
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

Manager Licensing
Limerick Generating Station
Exelon Generation Company, LLC
200 Exelon Way, KSA 3-E
Kennett Square, PA 19348

Assistant General Counsel
Exelon Generation Company, LLC
200 Exelon Way
Kennett Square, PA 19348

Correspondence Control Desk
Exelon Generation Company, LLC
P.O. Box 160
Kennett Square, PA 19348

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Limerick Generating Station
P.O. Box 596
Pottstown, PA 19464

Library
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

Director, Bureau of Radiation Protection
Pennsylvania Dept. of Environmental Protection
Rachel Carson State Office Building
P.O. Box 8469
Harrisburg, PA 17105-8469

Chairman
Board of Supervisors of Limerick Township
646 West Ridge Pike
Linfield, PA 19468

Dr. Judith Johnsrud
National Energy Committee
Sierra Club
433 Orlando Avenue
State College, PA 16803

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REQUEST FOR ADDITIONAL INFORMATION

REGARDING PROPOSED AMENDMENT REQUEST

FOR IMPLEMENTATION OF ALTERNATIVE SOURCE TERM

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

By letter dated February 27, 2004, as supplemented by letters dated October 25, 2004, and October 10, 2005, Exelon Generation Company, LLC submitted a request for an amendment to the Technical Specifications for Limerick Generating Station, Unit Nos. 1 and 2 (LGS). The amendment would allow for the use of an alternative source term (AST) in the LGS design-basis radiological accident analysis. The following questions refer mainly to the October 10, 2005, LGS response to the Nuclear Regulatory Commission (NRC) staff's Request for Additional Information (RAI) dated August 18, 2005. The NRC has determined that a response to the following questions is necessary in order for the staff to complete its review.

1. Calculation LM-0642 was revised as a result of the first set of NRC RAI questions. The maximum water volume of the sump was changed to a smaller volume that does not include the condensate storage tank (CST) volume, and is based upon a "best estimate value" (see C-7 of LM-0642). Regulatory Position 5.1.3, "Assignment of Numeric Input Values," of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," states:

    > The numeric values that are chosen as inputs to the analyses required by [Title 10 of the *Code of Federal Regulations*] 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose.

    Table 2.1 of calculation LM-0646 states that LGS conforms to this Regulatory Position.

    In the case of determining the amount of buffer needed, a larger sump volume is conservative. For a shielding calculation, a smaller sump volume typically yields a more conservative dose.

    The LGS Updated Final Safety Analysis Report (UFSAR) says that the CST is the preferred and normal source for operations and is used for dilution of the source term for shielding calculation and previous design basis assessments (see Appendix A). The NRC staff, therefore, requests the following:

    a. Not including the CST water volume in LM-0642 appears to be inconsistent with the operation and current design basis. Justify not including this source of water, or include this volume in the determination of the buffer used.

    b. Provide the dilution volume used to determine the source term for the determination of the reactor heat removal (RHR) system shine to the control room. Describe and justify the sources of water used.

2. Provide a justification for not including “high volume” and “low volume” purging in the loss-of-coolant accident (LOCA) dose calculation. Provide information that quantifies the historical amount of purging and what controls are or will be put into place to restrict the amount of purging.

3. Provide the reference for the iodine resuspension model and justify its application to ASTs. What experimental data has been used to verify the application of this methodology to AST analyses?

4. The NRC staff requests further clarification on the LGS response to question 13, contained in the October 10, 2005, letter to the NRC. The LGS LOCA analysis calculates deposition in two steamlines. The contents of both steamlines are essentially deadheaded against the closed main steamline isolation valves (MSIVs) with a small amount of leakage through the valves. The Cline model cited provides a temperature profile for the steamline versus time. The Cline steamline temperatures are higher than the drywell temperatures. Likewise, the reactor vessel temperature (the source of the leakage) will be higher than drywell temperature.

    a. Given that the postulated leakage is essentially stagnant in this volume and that leakage is in contact with the steamline and reactor vessel, justify using the drywell temperature to calculate the volumetric flowrate in the volume up to the first MSIV. Explain why the temperature of the contents of this volume would not be conservatively approximated by the steamline temperature.

    b. Question 13, referenced above, requested a justification of why the generic Cline assessment was applicable and conservative for LGS. The maximum temperature of Figure 7 of the Cline study is approximately 560 Kelvin (approximately 550 E F). Per UFSAR Tables 1.3-1 and 10.3-1, the steamline design temperature is 582 E F. Page 17 of LM-0646 states that LGS conforms to Regulatory Position 5.1.3, which states that numeric values are chosen as inputs with the objective of determining a conservative postulated dose. Since using the design value yields a higher dose, justify the value chosen for the steamline temperature and the subsequent reductions in leakages at 24 and 96 hours.

5. LGS has requested a change in the assumption of mass released in a main steamline break. The current UFSAR Section 15.6.4.4 states that a release of 88,333 pounds of reactor water and 20,452 pounds of steam applies. LGS proposes to change this value to 140,000 pounds as provided in Standard Review Plan 15.6.4. Calculation LM-0644, Revision 1 uses the new value of 140,000 pounds to calculate the hemispherical radius of a steam cloud produced by this mass of steam. Calculation LM-0644 assumes that 40 percent of the mass flashes to steam or 56,000 pounds of steam is produced at 212 E F. This mass is similar to the mass of steam produced assuming 40 percent of the 88,333 pounds of reactor water flashes to steam and 20,452 pounds of steam in the steamline is released (i.e., 35,333 lbs of steam at 212 E F plus 20,452 pounds at 551.7 E F, the saturation temperature at 1060 psia, equals 55,785 pounds of steam). While the values released are very similar, the characteristics of the steam are different. The NRC staff requests a justification for the change in characteristics of the steam released since no change is being made to the design of the steam producing systems.

This justification and clarification are important since the thermodynamic properties of the cloud are used for the determination of the control room dose.

6. RG 1.183, Appendix D, states that if no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specifications. Table 2.1 of calculation LM-0644 states that the LGS analysis conforms to this Regulatory Position. Considering this conformance, please justify why the Cesium (Cs) source term for the main steamline break is based upon only the Cs attached to released iodine and not upon the Cs in the reactor coolant activity.

7. Attachment C of calculation LM-0646, Revision 1 provides a “Re-analysis of External Source Gamma Shine to Control Room.” The NRC staff requests the following clarifications concerning this analysis.

    a. For the control room shine dose calculation, several pipes (gamma sources) are not modeled based upon being directly behind other pipes. While the dose points in the horizontal plane (which bisects these pipes) may be shielded by the closest pipe to the control room, those dose points in the line of sight of both pipes may be non-conservative estimates by not including these pipes in the dose estimates. Please provide additional justification for not including the shielded pipes in the dose estimate (include clearer drawings to show the locations of line sources that are in the room adjacent to the control room). Confirm that excluding these pipes does not change the dose estimates at the worst location on the control room wall.

    b. Attachment C, Section 3.3, “Reactor Enclosure Cloud Shine,” describes the shielding analysis performed. Please provide additional detail of the analysis performed and a justification for the assumptions made. The existing description states that an extremely conservative gamma energy of 0.8 MeV is used. Please justify this value.

    c. UFSAR Table 15.6-22 states that whole-body dose for the LOCA included piping and containment shine. Please provide the current control room whole-body dose due to containment shine and compare it to the proposed new value. Describe any differences in the analyses performed.

    d. Attachment C, Section 3.1.2, states under “Source Terms” that only the non-noble gas release fractions are considered. Please confirm that for the analysis performed for Section 3.3 that the noble gases and all the RG 1.183, Table 1 release fractions are considered for the containment (primary and secondary) shine calculations. If not, please justify why they are excluded.

8. The RAI response dated October 10, 2005, to NRC question 12b states that, for the fuel handling accident (FHA), an artificial one air change per minute envelopes all possible control room ventilation and related power supply conditions. Please describe the analysis used to justify this assumption, and confirm that it provides the most limiting doses for the LGS control room under all possible design basis scenarios where it is used. In this confirmation, include the results of the sensitivity analysis that perform this confirmation.

9. The RAI response dated October 10, 2005, to NRC question 21 states that the unfiltered inleakage for ingress and egress considered in the original submittal and the UFSAR is no longer considered in the new analysis because of the presence of a door seal. The NRC staff requests further clarification on this change in assumptions. The staff understands that this door seal will be erected to prevent radiation from entering the main control room directly from the turbine enclosure.

    a. Please provide a further explanation of the door seals and the circumstances under which they are erected.

    b. Please describe how much time is needed to erect the door seals and whether this time is included in the control room dose calculation. If this time is included in the control room dose calculation, justify why it is excluded.

    c. The 10-cfm value for unfiltered inleakage from ingress and egress can occur from any doors into the control room. When the door seal is erected, how will essential personnel enter and exit the control room? If these personnel will enter from other doors, justify why the 10 cfm is not included to account for ingress and egress through these doors.

10. Calculation LM-0645, Revision 1 states the following:

    > The exhaust point under the assumed no filtration condition is the Reactor Building South Stack as per Ref. 12. This release point results in specific dispersion characteristics which are defined by unique dispersion factors, or χ/Q's, as derived in Ref. 6. The North Stack, which is used for releases filtered by the SGTS [standby gas treatment system], is located closer to the Control Room intake and therefore has higher χ/Q's, as also derived in Ref. 6. However, the SGTS is designed to remove at least 99% of the iodine that would otherwise be released; this filtration more than overcomes the effect of the higher χ/Q's, as demonstrated herein, so the South Stack release unfiltered is bounding.

    RG 1.183, Regulatory Position 5.1.2 states:

    > Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures.

    LGS has requested to remove the operability requirement during refueling for the SGTS in Technical Specification 3.6.5.3, but credits this system in the FHA analysis. Given that the SGTS is not required to be operable during fuel handling, and that LGS states it conforms to Regulatory Position 5.1.2, justify how the FHA dose calculation can credit the SGTS filtration or recalculate the FHA doses without credit for the SGTS.

11. In the August 18, 2005, letter from T. Tate (NRC) to Christopher M. Crane (Exelon), the NRC staff requested a confirmation that the full power conditions are most limiting or

provide justification for why other conditions were not evaluated to determine the most-limiting release conditions (Question #8). The staff is unclear whether the response given in October 10, 2005, provides the requested confirmation. Please clarify your response. Do the full power conditions analyzed in the submittal provide the most limiting radiological doses?

APPENDIX A

UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

REFERENCES FROM REQUEST FOR ADDITIONAL INFORMATION

UFSAR Section II.B.2 states:

> The RHR system recirculates reactor water when it operates in the shutdown cooling mode. Before operation in this mode can be initiated, the reactor must be depressurized to less than 75 psig. This depressurization is expected to remove substantially all of the noble gases released into the reactor water. Following an accident, the HPCI [high-pressure coolant injection], RCIC [reactor core isolation coolant], RHR (LPCI mode), and CS systems would inject water into the RCS [reactor coolant system]. **This water from the condensate tank and/or the suppression pool would dilute the reactor water prior to the initiation of shutdown cooling with the RHR system.** This review assumed that there are no noble gases in the reactor water in the RHR system for the shutdown cooling mode and that the reactor water is diluted by the suppression pool water volume.

UFSAR Section II.K.3.22, "Automatic Switch-Over of RCIC System Suction - Verify Procedures and Modify Design," states:

> Modifications have been made to change the RCIC system suction valve logic to **automatically switch suction from the CST to the suppression pool on low CST level**.

UFSAR Section 3A.3.2.3, "Design Basis Accident" states:

> Shortly after a DBA, the ECCS [emergency core cooling system] pumps (HPCI, CS, and LPCI) **automatically start pumping CST water or suppression pool water** in to the RPV.

UFSAR Section 6.3.2.2.1, "High Pressure Coolant Injection System" states:

> The **normal alignment of the HPCI system initially injects water from the CST** instead of water from the suppression pool. An alternate alignment to the suppression pool is also available during periods when the CST is not available.

UFSAR Section 7.3.1.1.1.1.3, "HPCI Initiating Circuits" states:

> The **preferred source** of water for the HPCI is the CST.

UFSAR Section 7.4.1.1.3.6, "RCIC Actuated Devices" states:

> Three pump suction valves are provided in the RCIC system. One valve (F010) lines up pump suction from the CST; the other two (F029, F031) do so from the suppression pool. **The CST is the preferred source.**