Mr. Charles G. Pardee
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 - REQUEST FOR ADDITIONAL INFORMATION RELATED TO ALTERNATIVE SOURCE TERM LICENSE AMENDMENT REQUEST (TAC NOS. ME0068 AND ME0069)

Dear Mr. Pardee:

By letter to the Nuclear Regulatory Commission (NRC) dated October 23, 2008 (Agencywide Documents Access and Management System Accession No. ML083100149), Exelon Generation Company, LLC, submitted a request to replace the current accident source term used in design-basis radiological analysis with an alternative source term pursuant to Title 10 of the *Code of Federal Regulations* Part 50.67, “Accident Source Term,” for the LaSalle County Station, Units 1 and 2.

The NRC staff is reviewing your submittal and has determined that additional information is required to complete the review. The specific information requested is addressed in the enclosure to this letter. This information was previously requested on October 23, 2009, and November 6, 2009. Due to the large amount of calculations that must be redone, we understand that you were not able to answer these questions within a 30 day response period. During a discussion with your staff on January 25, 2010, it was agreed that you would provide a response by April 2, 2010.

The NRC staff considers that timely responses to requests for additional information help ensure sufficient time is available for staff review and contribute toward the NRC’s goal of efficient and effective use of staff resources. Please note that if you do not respond to this letter within the prescribed response time or provide an acceptable alternate date in writing, we may reject your application for amendment under the provisions of Title 10 of the *Code of Federal Regulations*, Section 2.108. If circumstances result in the need to revise the requested response date, please contact me at (301) 415-3719.

Sincerely,

Cameron S. Goodwin, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosure:
Request for Additional Information

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/RA/

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REQUEST FOR ADDITIONAL INFORMATION

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

In reviewing the Exelon Generation Company's (Exelon's) submittal dated October 23, 2008 (Agencywide Documents Access and Management System Accession No. ML083100149), related to the request to replace the current accident source term used in design-basis radiological analysis with an alternative source term pursuant to Title 10 of the *Code of Federal Regulations* Part 50.67, "Accident Source Term," for the LaSalle County Station (LSCS), Units 1 and 2, the Nuclear Regulatory Commission (NRC) staff has determined that the following information is needed in order to complete its review:

Accident Assessment Dose Branch

1. In its September 28, 2009, request for additional information (RAI) response, Exelon referenced information previously provided, as well as providing additional information about meteorological data processing, wind speeds and atmospheric stability at LaSalle and several other northern Illinois nuclear power plant sites. The response noted that meteorological data at the other sites were more similar to each other than to LaSalle. In addition, Exelon's response noted the meteorological tower at LaSalle had been moved between when historical measurements made in the 1970s, and when the 1998 through 2003 data were collected. Exelon attributed the differences in the data to the location of the current tower in open homogenous terrain.

   After reviewing Exelon's response, the NRC staff noted that wind measurements reported for LaSalle in 1998 appear to average about one mile per hour lower, and the occurrence of unstable conditions, while low were higher in 1998 than between 1999 through 2003. Also, for temperature difference measurements between 114.3 and 10.1 meters above ground level, extremely unstable conditions (stability class "A") were reported to occur 26 times in 1998, and only a total of two times in the remaining five years between 1999 through 2003. These variations within the data raise concerns for the NRC staff regarding the accuracy of the data and its characterization of site conditions. Therefore, please provide a description of any modifications to the meteorological measurement program in the 1998 through 2003 time frame, and the potential impacts that the changes could have on the reported data. Identify which portions of each measurement system received routine calibrations and confirm that conversion factors (e.g., conversion to degrees Centigrade per 100 meters), or other data modifications/processes were correctly applied. Finally, please provide any other information that supports that the 1998 through 2003 data are of high quality.

2. For calculation of ground level release exclusion area boundary (EAB) and low population zone (LPZ) atmospheric dispersion factors (X/Q values), Exelon stated in its September 28, 2009, RAI response that use of meteorological data based upon delta temperature measurements between the 114.3 and 10.1 meter levels was appropriate because the Reactor Building complex would generate aerodynamic dispersive effects

within that height interval. Therefore, the 114.3 to 10.1 meter height measurements were assumed to be more representative than data based upon the lower, 61.0 to 10.1 meter measurements.

Regulatory Guide (RG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," states that atmospheric stability should be determined by vertical temperature difference between the release height and the 10 meter level. NRC staff notes that equations in the PAVAN computer code used to calculate the X/Q values include a factor to make an adjustment for aerodynamic dispersive effects, independent of atmospheric stability. Therefore, NRC staff judges that data input into the PAVAN code should use measurements between the release height and 10-meter level, in this case the 61.0 to 10.1 meter measurements. NRC staff has made scoping calculations using data based upon temperature measurements between 61.0 and 10.1 meters above ground level and found the results more limiting than when using the data based upon the 114.3 to 10.1 meter measurements. Therefore, please justify why X/Q values based upon the 114.3 and 10.1 meter temperature measurements should be used in the dose assessment in support of the current license amendment request and should become the new ground level EAB and LPZ licensing basis X/Q values.

3. With regard to an assumed diffuse release through the secondary containment wall following a fuel-handling accident (FHA), the analyses does not assume that the activity was homogeneously distributed as specified in RG 1.194. Instead, according the submittal, the analysis assumes that the effluent is contained in a one cubic foot volume, which is considered to be conservative by the licensee. The licensee also noted that potential leakage through the wall was expected to be small, with little driving force, and that there are no penetrations through the wall. Thus, the use of the diffuse release assumption is conservative.

   RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” states that:

   “Diffuse source modeling should be used only for those situations in which the activity being released is homogenously distributed throughout the building and when the assumed release rate from the building surface would be reasonably constant over the surface of the building.”

   NRC staff has determined that application of the diffuse release assumption is inappropriate given the guidance in RG 1.194, and that the licensee's application, as described above, could result in underestimation of the resultant X/Q values. Therefore, the NRC staff requests that the licensee justify how the integrity of the secondary containment wall can be maintained to prevent a release through the wall or provide an alternative release scenario and justification to assess release of the effluent to the environment in some other way.

4. In regards to your FHA analysis, please provide a comprehensive list of all analyzed and unanalyzed secondary containment potential release points and pathways to the environment and control room. These pathways should include those pathways to

adjacent buildings that could leak to the environment or to the control room (i.e. Secondary Containment heating, ventilation and air conditioning, ductwork, structural openings, etc.). Additionally, the licensee's evaluation of the pathways should consider the effects of inoperability of other safety systems such as the Secondary Containment Isolation Valves and Standby Gas Treatment System (SGTS). For each potential release point or pathway, the licensee should provide the following:

a. The results of its dose analysis demonstrating that 10 CFR 50.67 regulatory limits are met;

b. If a dose analysis has not been performed, a technically sound basis for why this release point or pathway is bounded by other analyzed release points; and

c. An explanation for how the existing proposed TS changes or, as necessary, new revised TSs will ensure that the dose limits are met.

5. In Attachment 4, “Summary of Regulatory Commitments,” of the October 23, 2008, supplement, the licensee states that no credit is taken for filtration by the SGTS. However, in the same supplement, the licensee provides a commitment to provide "…prompt methods…to enable ventilation systems to draw the release from a postulated FHA in the proper direction such that it can be treated and monitored." This appears to be a reference to the SGTS. Based on the proposed technical specification (TS) changes from the license amendment request (LAR), it does not appear that the SGTS will be required to be operable during non-recently irradiated fuel handling operations (i.e., after the reactor has been subcritical for 24 hours). As such, it would not be available for performing the safety function described in the commitment. With the SGTS inoperable, the licensee would be unable to create the differential pressure inside the secondary containment necessary for the purposes of directing the radioactive release from a FHA through the SGTS filtration and the Main Stack. As such, other potential release points or pathways, such as a smaller secondary containment penetrations and pathways the licensee has determined must remain closed even after 24 hours of decay time that have not been analyzed or considered, may become relevant dose contributors. Therefore, the staff requests the licensee to provide the following information:

    a. An analysis demonstrating that the SGTS can perform its safety function under all possible plant configurations related to secondary containment operability during fuel handling operations. This analysis should consider the potential impacts of differential pressures caused by local wind conditions.

    b. Appropriate TSs related to SGTS operability during periods when it is credited for performing a safety function related to mitigating the consequences of a FHA.

6. In the licensee's October 23, 2008, supplement letter, Attachment 5, Table 1 “Regulatory Guide 1.183 Conformance Matrix,” states that the submittal “conforms” to Regulatory Position 5.1.2 and that “No credit is taken for mitigation factors for the FHA analysis.” However, in Attachment 9 of the October 23, 2008, supplement, the licensee determined that one of the six analyzed release pathways could not be opened during movement of irradiated fuel even after a 24-hour decay period as proposed in the LAR.

By proposing a limited number of acceptable penetrations and openings that can be breached, the licensee credits the capability of any remaining secondary containment accident mitigation features as being capable of performing their safety functions for the analyzed conditions for the duration of their mission times. However, the licensee's proposed TS changes remove all requirements for all secondary containment accident mitigative features after 24 hours.

Therefore, the NRC staff requests that the licensee provide revised TS changes consistent with its proposed revised analysis of record, that ensure the lowest functional capability or performance level of equipment credited for functioning or actuating to mitigate the design basis FHA.

7. Based on the differences identified between the licensee's analyses and its statement regarding conformance with Regulatory Position 5.1.2 discussed in the above question 6, the NRC staff requests that the licensee reevaluate its conformance with Regulatory Position 5.1.2 and provide additional justification that all credited accident mitigation features are classified as safety-related, are required to be operable by TSs, are powered by emergency power sources, and are either automatically actuated, or in limited cases, have actuation requirements explicitly addressed in emergency operating procedures.

8. The NRC staff requests that the licensee explain how the monitoring of radioactive releases resulting from a FHA or “inadvertent release of radioactive material” (General Design Criteria 63, “Monitoring fuel and waste storage,” and 64, “Monitoring radioactivity releases”) will be accomplished with the secondary containment open. The current licensing basis for LaSalle assumes the secondary containment is operable during fuel handling operations, and by extension, would be operable during a FHA or an “inadvertent release of radioactive material.” As such, any radiation monitoring and filtering equipment inside secondary containment would have been designed, located, and calibrated based on the current design and licensing basis. The proposed changes could impact the effectiveness of that monitoring equipment. For example, the timing of proceduralized operator actions related to indications or alarms from this equipment potentially could be delayed or prevented by a reduced effectiveness of this equipment. The NRC staff believes the ability to effectively monitor the radioactive release is critical to the protection of the public and plant personnel.

<u>Containment and Ventilation Branch</u>

<u>Background:</u>

The Containment and Ventilation Branch questions presented in the August 27, 2009, RAI attempted to establish that sufficient concentration of sodium pentaborate would reach the suppression pool to buffer the pool pH to greater than 7.0 within the time limit established by the accident analysis. In their September 28, 2009, response, Exelon Generation Company (EGC) indicated that emergency core cooling system (ECCS) subsystems; High Pressure Core Spray (HPCS), Low Pressure Core Spray (LPCS), and Residual Heat Removal (RHR), would be in use during a loss-of-coolant accident (LOCA). The HPCS and LPCS systems inject above the core.

Updated Final Safety Analysis Report (UFSAR) Rev. 17, Figure 6.3-8, “Residual Heat Removal System,” show the RHR injection into the reactor vessel can be through the reactor recirculation pump discharge line (Low Pressure Core Injection mode) or directly to the reactor vessel at multiple locations above the core. RHR injection through the recirculation pump discharge will flow through the jet pumps on the non-broken loop to the reactor lower plenum. An attachment to an e-mail dated October 26, 2009, indicates that flow is down through the core, up through the jet pumps, and out through the recirculation loop suction. The UFSAR states that RHR can be used for other functions such as containment spray or suppression chamber spray. Attachment 8 to the amendment request (Calculation L-003064, “Suppression Pool pH Calculation for Alternative Source Terms, Revision 1”) states the flow will be up through the core. The October 28, 2009, e-mail also stated that low pressure core injection (LPCI) would be used. Calculation L-003064 shows that the minimum amount of sodium pentaborate will be injected into the lower reactor plenum over a 2-hour period. The sodium pentaborate flow of 41.2 gallons per minute is diluted in the lower plenum with roughly 20,000 gallons of water plus an undetermined ECCS flow from above the core shroud and core. RAI-3 and RAI-4 (August 27, 2009, letter), tried to determine how LSCS would verify that adequate sodium pentaborate actually reached the suppression pool in the required time limit. EGC responded in the September 28, 2009, letter, that analysis of the injection time and standby liquid control (SLC) storage tank capacity and sodium pentaborate concentration provides the verification requested.

The dose analysis is based on sufficient buffering of the suppression pool. According to RG 1.183, if the suppression pool pH is maintained greater than 7.0 no re-volatilization of the radioiodines is assumed. According to the EGC analysis, LSCS has 3.5 hours to buffer the suppression pool. Their analysis also indicates that they take 2 hours to empty the SLC tank into the reactor. The SLC injects below the core plate. The design-basis loss-of-coolant break location is above the core plate/shroud assembly. According to the material provided by EGC, flow from the lower reactor plenum is both up through the core and by reverse flow up through the jet pumps. While ECCS flow is down through the core and down through the jet pumps via the LPCI system.

1. Provide a plant-specific calculation or analysis or a topical report calculation or analysis documenting how is stratification of the sodium pentaborate injected into the lower reactor plenum prevented and mixing is verified.

2. Assuming adequate mixing is assured by Question 1, what mass of sodium pentaborate reaches the suppression pool at the end of the SLC injection period? Include in the evaluation further dilution of the sodium pentaborate solution mixing with ECCS water that enters the core shroud/reactor vessel annulus.

3. Assuming a continuous dilution of sodium pentaborate concentration in the lower reactor plenum by the ECCS, what is the mass of sodium pentaborate reaching the suppression between the end of SLC injection and the 3.5-hour time limit? Include in the evaluation further dilution of the sodium pentaborate solution mixing with ECCS water that enters the core shroud/reactor vessel annulus.