UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 1, 2011

Mr. Michael J. Pacilio
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 – CLARIFICATION OF INFORMATION CONTAINED IN NRC SAFETY EVALUATION FOR ALTERNATIVE SOURCE TERM LICENSE AMENDMENT (TAC NOS. ME5664 AND ME5665)

Dear Mr. Pacilio:

By letter dated September 6, 2010, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML101750625), the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 197 to Facility Operating License No. NPF-11 and Amendment No. 184 to Facility Operating License No. NPF-18 for the LaSalle County Station (LSCS), Units 1 and 2, respectively. The amendments revised the Technical Specifications to support the application of alternative source term methodology with respect to the loss-of coolant accident and the fuel handling accident in response to Exelon Generation Company, LLC (EGC, the licensee) application dated October 23, 2008, as supplemented by letters dated September 28, November 18, 2009, March 29 and August 3, 2010 (ADAMS Accession No. ML083100153, ML092710196, ML093220838, ML100890060 and ML102230205), respectively.

Subsequently, by letter dated November 18, 2010 (ADAMS Accession No. ML103230113) EGC identified the need to clarify information documented in the NRC safety evaluation (SE) related to the classification of the Standby Liquid Control (SLC) system at LSCS as discussed below.

The fourth paragraph of page 4 of the SE states:

> . . . the licensee indicated that the SLC system at LSCS was classified as a safety-related system.

Additionally, the fifth paragraph of page 9 of the SE states:

> LSCS states that the SLC system is a safety-related system and meets the following requirements . . .

The LSCS Updated Final Safety Analysis Report describes the SLC system as a special safety system in Section 9.3.5.1 and Section 9.3.5.3. EGC stated in the October 23, 2008 submittal that the SLC system is a safety related system; however, clarifying information regarding the classification of the SLC system was provided in the Attachment of the November 18, 2009 letter that stated:

> The SLC system components are currently classified as augmented quality with special requirements, and as such, these components are treated as safety related for design, procurement, and maintenance/testing to meet the requirements of 10 CFR 50, Appendix B, "Quality Assurance Program," and General Design Criterion 4.
>
> The devices were originally procured, installed, and maintained as environmentally qualified to the requirements of 10 CFR 50.49. The components are currently classified as "augmented quality" and will continue to be treated as safety related and thus, will ensure that the environmental criteria are met.

The SLC system components are treated as safety related for design, procurement, and maintenance/testing purposes. Additionally, the classification of the SLC system as augmented quality, in lieu of safety related has no impact on the SLC system's ability to support the application of alternative source term methodology with respect to the loss-of coolant accident and the fuel handling accident.

Enclosed are the revised versions of pages 4 and 9 of the SE. The incorrect information was editorial in nature. The line in the right margin indicates the areas of correction. The correction does not affect the NRC staff's conclusions associated with the SE. Please substitute the revised pages of the SE for the ones originally provided. We regret any inconvenience this may have caused you.

Sincerely,

Araceli T. Billoch Colón

Araceli T. Billoch Colón, 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:
Revised Safety Evaluation

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Attachment 1 to Reference 1 that the main steam drain lines are not credited for deposition and hold up of any MSIV leakage for the purposes of the LOCA re-analysis in the proposed AST.

In support of the deletion of their LCS, LSCS did not utilize the generic methodologies presented in Reference 3, which focused on the use of an earthquake experience database, to seismically qualify ALT pathway components. Instead, the licensee opted to perform a plant-specific method of qualification, utilizing a complete analytic evaluation of the seismic adequacy of the ALT pathway piping, connected components, the main condenser, and supports. Seismic walkdowns were performed in support of the seismic ruggedness demonstration to identify conditions of piping and support configurations which would have resulted in seismically-induced pressure boundary failure and fission product release from the main steam line piping and main stream line drain piping. Additionally, the licensee analytically demonstrated the seismic adequacy of the turbine building, based on the fact that it contained many of the piping runs and pipe supports included in the ALT pathway. As previously stated, the NRC staff concluded that the methodology utilized by the licensee, including the analytical evaluations and seismic walkdowns, for determining the seismic ruggedness of the ALT pathway at LSCS was found acceptable (Reference 4). Additionally, the licensee confirmed, in response to the NRC staff's request for additional information (RAI) in Reference 2, that no modifications to the ALT pathways would be necessary to support the implementation of the proposed AST implementation. Thus, no changes are needed to the conclusions surrounding the seismic adequacy of the ALT pathway.

Based on the conclusion by the NRC staff in Reference 4 that the ALT pathways utilized at LSCS were demonstrated to be seismically adequate and no modifications to the ALT pathways are necessary to support the proposed AST implementation, the NRC staff concludes that the licensee has provided reasonable assurance that the ALT pathways are capable of performing their safety function during and following an SSE.

2.1.2.2 Standby Liquid Control (SLC) System Seismic Evaluation

In performing the re-evaluation of the LOCA DBA, the licensee indicated in Attachment 1 to Reference 1 that credit would be taken for controlling the pH in the suppression pool following a LOCA by injecting sodium pentaborate into the reactor core, utilizing the SLC system. Detailed design information regarding the SLC system at LSCS can be found in Section 9.3.5, "Standby Liquid Control System (BWRs)," of the facility's Updated Final Safety Analysis Report (UFSAR). To demonstrate that the SLC system is capable of performing its intended safety function during a LOCA following AST implementation, the licensee addressed the guidance provided by the NRC staff in Reference 5. This guidance provides four review guidelines which the licensee should use to evaluate whether their plant-specific SLC system can be credited as either safety-related or comparable to a safety-related system for the purposes of controlling the pH of the reactor coolant following a LOCA. Review guidelines 1 and 4 are addressed below, while the remaining guidelines are addressed in Section 2.3.2 of this safety evaluation report.

With respect to the first guideline found in Reference 5, the licensee indicated that the SLC system at LSCS was classified as a special safety-related system. Given this classification, the SLC system at LSCS is designed to Seismic Class I requirements based on the seismic qualification methodologies found in Chapter 3, "Design of Structures, Components, Equipment, and Systems," of the LSCS' UFSAR. Additionally, the licensee referenced the redundancy of the active components within the SLC system, which is addressed by the fourth guideline of

2.3.2 TS Section 3.1.7, "Standby Liquid Control (SLC) System"

The proposed change revises the applicability of TS Section 3.1.7 to add the requirement for the limiting condition for operation (LCO) to be met in Mode 3. This change implements AST assumptions regarding the use of the SLC system to buffer the suppression pool following a LOCA involving significant fission product release. The proposed change revises Condition C Required Actions to add an additional requirement, C.2, to be in Mode 4 with a completion time of 36 hours.

LSCS evaluated the suppression pool pH over the 30-day duration of the DBA LOCA and demonstrated that with injection of sodium pentaborate through the SLC system, the pH will remain above 7.0. Therefore, iodine conversion to elemental with re-evolution is considered inconsequential in the LOCA calculation. The control of pH also significantly limits the potential for airborne release from sub-cooled ECCS leakage inside and outside of secondary containment.

LSCS proposes 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 (Le., pH control post-LOCA).

The SLC system consists of the boron solution tank, the test water tank, two positive displacement pumps, two explosive valves, two motor-operated pump suction valves, and associated local valves and controls are located in the secondary containment. The liquid is piped into the reactor vessel and discharged near the bottom of the core shroud.

LSCS stated that the SLC system meets the following requirements:

- The SLC system is provided with standby alternating current (ac) power supplemented by the emergency diesel generators.
- The SLC system is seismically qualified in accordance with RG 1.29, "Seismic Design Classification," and Appendix A to 10 CFR Part 100.
- The SLC system is incorporated into the plant's American Society of Mechanical Engineers Code inservice inspection and inservice testing programs based upon the plant's code of record (10 CFR 50.55a).
- The SLC system is incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants."
- The SLC system meets 10 CFR 50.49, "Environmental qualification of electrical equipment important to safety for nuclear power plants," and Appendix A (GDC 4, "Environmental and Dynamic Effects Design Bases") to 10 CFR Part 50.

The LSCS SLC system activation steps are in a safety-related plant procedure (Le., Emergency Operating Procedure (EOP) LGA-001, "RPV Control"). LSCS states that they will revise

> The SLC system components are currently classified as augmented quality with special requirements, and as such, these components are treated as safety related for design, procurement, and maintenance/testing to meet the requirements of 10 CFR 50, Appendix B, "Quality Assurance Program," and General Design Criterion 4.
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Sincerely,

/RA/

Araceli T. Billoch Colón, 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:
Revised Safety Evaluation

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