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RS-13-092 10 CFR 50.90

September 5, 2013

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

LaSalle County Station, Units 1 and 2
Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Subject: License Amendment Request to Revise Peak Calculated Primary Containment Internal Pressure

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change would revise the peak calculated primary containment internal pressure for the design basis loss-of-coolant accident (LOCA) described in TS 5.5.13, "Primary Containment Leakage Rate Testing Program." The peak calculated primary containment internal pressure, $P_a$, is increased from 39.9 psig to 42.6 psig. The proposed increase in $P_a$ is the result of a LOCA-Drywell Temperature sensitivity analysis performed by General Electric Hitachi (GEH).

Plant operations in TS 5.5.13 are currently administratively controlled under the provisions of NRC Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," to assure that plant safety is maintained. This license amendment request is submitted in accordance with the guidance in AL 98-10. In accordance with the guidance of AL 98-10, EGC submits the proposed change as a required license amendment request to resolve a non-conservative TS. As such, this is not a "voluntary request from a licensee to change its licensing basis" and should not be subject to "forward fit" considerations.

The attached request is subdivided as follows:

- Attachment 1 provides an evaluation of the proposed change.
- Attachment 2 provides the current TS page with the proposed change indicated.

The proposed amendment has been reviewed by the LSCS Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed license amendment by September 5, 2014. Once approved, the amendment will be implemented within 60 days.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," paragraph (b), EGC is notifying the State of Illinois of this application for license amendment by transmitting a copy of this letter and its attachments to the designated State Official.

There are no regulatory commitments contained in this letter. Should you have any questions concerning this letter, please contact Ms. Lisa A. Simpson at (630) 657-2815.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 5th day of September 2013.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachments:
1) Evaluation of Proposed Change
2) Markup of Technical Specifications Page

cc: NRC Regional Administrator, Region III
NRC Senior Resident Inspector, LaSalle County Station
Illinois Emergency Management Agency – Division of Nuclear Safety

**ATTACHMENT 1**
**Evaluation of Proposed Change**

Subject: License Amendment Request to Revise Peak Calculated Primary Containment Internal Pressure

## 1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station, Units 1 and 2.

Exelon Generation Company, LLC (EGC) proposes to revise the peak calculated primary containment internal pressure for the design basis loss-of-coolant accident (LOCA) described in Technical Specifications (TS) 5.5.13, "Primary Containment Leakage Rate Testing Program." The proposed change would revise the peak calculated primary containment internal pressure, $P_a$, from 39.9 psig to 42.6 psig.

The proposed increase in $P_a$ is the result of a LOCA-Drywell Temperature sensitivity analysis performed by General Electric Hitachi (GEH). The GEH sensitivity analysis (2012 GEH Analysis, Reference 1) indicated that $P_a$ is higher when using an initial lower drywell temperature (98 °F) versus the maximum allowable drywell temperature (135 °F). The proposed increase also incorporates various secondary issues affecting the analysis of record (AOR) (Reference 2). The result of utilizing a drywell temperature input of 98 °F and the effect of the secondary issues is a $P_a$ of 42.6 psig.

EGC requested GEH to perform the sensitivity analysis in support of the LSCS Extended Power Uprate (EPU) project. On June 11, 2013, EGC announced the cancellation of its LSCS EPU project; however, the cancellation of EPU does not negate the need for the proposed change.

In accordance with NRC Administrative Letter (AL) 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," (Reference 3), administrative controls implemented under EGC Operability Evaluation OE 11-002 (Reference 4) assure that plant safety is maintained. In accordance with the guidance of AL 98-10, EGC submits the proposed change as a required license amendment request to resolve a non-conservative TS. As such, this is not a "voluntary request from a licensee to change its licensing basis" and should not be subject to "forward fit" considerations.

Approval of this amendment application is requested by September 5, 2014. Once approved, the amendment will be implemented within 60 days.

## 2.0 DETAILED DESCRIPTION

TS 5.5.13.b currently states:

> "The peak calculated primary containment internal pressure for the design basis loss of coolant accident, $P_a$, is 39.9 psig."

The proposed change would revise TS 5.5.13.b by replacing the $P_a$ value of 39.9 psig with a value of 42.6 psig.

The revised TS 5.5.13.b would state as follows:

> "The peak calculated primary containment internal pressure for the design basis loss of coolant accident, $P_a$, is 42.6 psig."

Attachment 2 provides mark-ups of the affected TS pages for the proposed change.

## 3.0 BACKGROUND

Peak containment internal pressure during a LOCA is calculated to determine, in part, whether the design pressure limit for the containment building would be exceeded during a design basis accident. LSCS TS 5.5.13, "Primary Containment Leakage Rate Testing Program," describes the peak calculated primary containment internal pressure during the design basis LOCA, $P_a$.

A LOCA is initiated by the rupture of the primary coolant system piping. The primary coolant flashes to steam and escapes through the pipe break. As the steam is released to containment, containment atmosphere pressure and temperature quickly increase. During a LOCA event, the initial blow down of the primary coolant system adds mass and energy to the containment.

The current LSCS AOR peak calculated primary containment internal pressure, $P_a$, is 39.9 psig, as stated in the LSCS UFSAR and TS (Reference TS 5.5.13 and UFSAR 6.2.1.1.3). Results of the 2012 GEH Analysis concluded that the peak calculated primary containment internal pressure, $P_a$, is 42.6 psig, which is 2.7 psig higher than the value determined in the AOR (i.e., 42.6 psig – 39.9 psig = 2.7 psig). This proposed increase to $P_a$ includes both the effects of using the lower drywell temperature and the secondary issues regarding the AOR.

The proposed increase in $P_a$ is the result of a LOCA-Drywell Temperature sensitivity analysis performed by GEH (2012 GEH Analysis) in support of the LSCS EPU project. The 2012 GEH Analysis indicated that $P_a$ is higher when using an initial lower drywell temperature (98 °F) versus the maximum TS allowable drywell temperature (135 °F). The effect of using the lower drywell temperature increases $P_a$ by 1.3 psig.

During the process of GEH conducting and EGC reviewing the analysis, additional secondary issues were identified regarding the AOR that also impact $P_a$. These issues include the Main Steam Safety Relief Valve (SRV) setpoints, the closure time for the Main Steam Isolation Valves (MSIVs), the core flow conditions permitted by the Power/Flow Map, and the reactor power level. The new Analysis corrects these issues. GEH has addressed each issue through their corrective action program. The cumulative effect of these secondary issues increases $P_a$ by 1.4 psig.

Although EGC canceled the EPU project for LSCS, this does not negate the need for the proposed change. In accordance with the guidance of AL 98-10, EGC submits the proposed change as a required license amendment request to resolve the non-conservative TS.

## 4.0 TECHNICAL EVALUATION

The proposed increase to $P_a$ incorporates an increase resulting from using the lower drywell temperature and an increase resulting from secondary issues regarding the AOR.

Initial Drywell Temperature

The current LSCS containment AOR (GE Project Task Report GE-NE-A1300384-02-01-R3, which is documented in Design Analysis L-002874) used an initial drywell temperature of 135 °F, which is the maximum value allowed by TS (Reference TS 3.6.1.5). The 2012 GEH Analysis indicated that the peak calculated primary containment internal pressure is higher when using an initial lower drywell temperature (98 °F) versus the maximum TS allowable drywell temperature (135 °F).

The 2012 GEH Analysis uses an initial drywell temperature of 98 °F based upon historical bulk average drywell temperature data for the time period of June 1, 2007, through March 22, 2011. The evaluation examined the bulk average drywell temperature in one hour increments to ensure that time of the day and operating variations were captured. The evaluation included 2 sigma standard deviation and found that when the plant operates at ≥ 90% power level, the drywell temperature will be ≥ 100 °F. The drywell temperature of 98 °F was selected to add conservatism.

The use of the TS limiting drywell temperature as input for the containment analysis is conservative in the determination of the peak drywell temperature; however, the use of the maximum TS allowable drywell temperature is not conservative in relation to the peak calculated primary containment internal pressure, $P_a$. The effect of using the lower initial drywell temperature increases $P_a$ by 1.3 psig.

Secondary AOR Issues

During the process of GEH conducting and EGC reviewing the analysis, additional secondary issues were identified regarding the AOR that also impacts $P_a$. Exelon requested that GEH evaluate the secondary issues used in the containment analysis to determine how they may affect $P_a$. GEH has addressed each of the secondary issues through their corrective action program. The 2012 GEH Analysis corrects these issues. The cumulative effect of these secondary issues increases $P_a$ by 1.4 psig. The issues identified during the review of the 2012 GEH Analysis include the following:

- Main Steam Safety Relief Valve (SRV) Setpoints

  The Main Steam SRV setpoints used in the 2012 GEH Analysis impact the $P_a$ results because the opening of the SRVs directs energy from the reactor pressure vessel (RPV) to the suppression pool instead of releasing the energy to the drywell through the break. Higher SRV setpoints result in the SRVs remaining closed longer, thus directing more energy to the drywell. The LSCS SRVs are designed to open in safety mode (i.e., as safety valves with spring mode of operation) or relief mode (i.e., a pneumatic piston/cylinder actuator providing the mode of operation). SRV operation in the safety mode has a higher opening setpoint than in the relief mode. Therefore, in order to provide a bounding calculated primary containment internal pressure, $P_a$, the 2012 GEH Analysis uses the higher Main Steam SRV spring setpoints credited in TS 3.4.4, "Safety / Relief Valves (S/RVs)." This differs from the AOR which used the lower SRV relief mode setpoints.

- Main Steam Isolation Valve (MSIV) Closure Time

  The MSIV closure time assumed in the 2012 GEH Analysis impacts the $P_a$ results since the longer time it takes the MSIVs to close, the longer RPV mass and energy is directed to the turbine. Conversely, a shorter MSIV closure time means that more RPV mass and energy will be released to the drywell via the break, resulting in a higher $P_a$. The AOR uses a MSIV closure time of 5 seconds, which is the maximum MSIV closure time allowed in TS Surveillance Requirement (SR) 3.6.1.3.6. In order to provide a bounding peak calculated primary containment internal pressure, the 2012 GEH Analysis uses a shorter MSIV closure time of 3.5 seconds, based on the minimum TS SR 3.6.1.3.6 MSIV isolation time with an assumed 0.5 second from break occurrence to initiation of MSIV closure.

- Core Flow Conditions Permitted by the Power/Flow Map

  Core flow conditions assumed in the analysis affect the mass and energy released to the drywell through the break. GEH evaluated plant operation at Rated Core Flow (RCF), Increased Core Flow (ICF), and Maximum Extended Load Limit Analysis (MELLA) conditions to ensure that the condition which maximizes the energy being released to the drywell and provides a bounding $P_a$ is accounted for. Additionally, feedwater temperature affects the energy released through the break, therefore both the Final Feedwater Temperature Reduction (FFWTR) and Normal Feedwater Temperature (NFWT) were considered. The 2012 GEH Analysis considered the different core flow and feedwater temperature conditions, and the results are based upon the conditions that yield the highest $P_a$.

- Reactor Power Level

  A lower reactor power level results in less energy being released to the drywell via the break, resulting in a lower $P_a$.

  The AOR states that the analyses were performed at a reactor power level of 3559 MWt. However, it was subsequently determined that the AOR analyses were actually performed at 3789 MWt. This discrepancy was addressed by GEH through their corrective action program.

  The 2012 GEH Analysis used a maximum reactor power level of 3559 MWt, which is 102% of the stretch power uprate licensed thermal power (3489 MWt). As a consequence of the decrease in reactor power level used in the 2012 GEH Analysis compared to the AOR, the impact of this specific change was a decrease in $P_a$.

<u>Summary of Changes to Peak Calculated Primary Containment Internal Pressure and Evaluation</u>

The current LSCS AOR peak calculated primary containment internal pressure, $P_a$, is 39.9 psig, as stated in the LSCS UFSAR and TS (Reference TS 5.5.13 and UFSAR 6.2.1.1.3). Results of the 2012 GEH Analysis concluded that the peak calculated primary containment internal pressure, $P_a$, is 42.6 psig, which is 2.7 psig higher than the value determined in the AOR (i.e., 42.6 psig – 39.9 psig = 2.7 psig). The proposed increase of 2.7 psig to $P_a$ incorporates an increase resulting from using the lower drywell temperature (i.e., 1.3 psig) and an increase resulting from secondary issues regarding the AOR (i.e., 1.4 psig).

As defined in LSCS UFSAR Table 6.2-1, the containment design pressure is 45 psig. Since the peak calculated primary containment internal pressure resulting from the non-conforming condition (42.6 psig) is less than the containment design pressure (45 psig), the increased $P_a$ has no effect upon the capability of the primary containment to accomplish its safety-related functions.

Containment leakage rate testing is addressed in UFSAR Section 6.2.6 and LSCS TS 5.5.13. The LSCS containment leakage rate testing program implements testing requirements in accordance with 10 CFR 50 Appendix J, Option B, as modified by any approved exemptions, and criteria contained in RG 1.163, "Performance-Based Containment Leak Test Program." Tests that measure containment isolation valve leak rates (Type C tests) are performed using the peak calculated primary containment internal pressure value of 42.6 psig.

As described in TS 5.5.13 and LSCS UFSAR Table 6.2-23, the test pressure used for containment leakage testing is based upon the $P_a$ of 39.9 psig. The higher calculated peak pressure $P_a$ will affect the containment leakage testing results. Leak rate testing based upon the higher $P_a$ of 42.6 psig would have increased the measured leakage. EGC Operability Evaluation OE 11-002 demonstrates that there is adequate margin to accommodate this increase.

As described in LSCS UFSAR 3.11.1.1.1, the environmental qualification of safety-related electrical components and instrumentation located in the primary containment are based upon the design pressure (45 psig), not the calculated $P_a$ pressure (39.9 psig). Hence, the revision of TS 5.5.13.b that increases the $P_a$ value to 42.6 psig has no effect upon environmental qualification.

The change in $P_a$ does not affect the LSCS analysis of radiological consequences of a LOCA with respect to radiological dose at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), Control Room (CR), or Technical Support Center (TSC). Calculated doses during a LOCA are dependent on the maximum allowable containment atmosphere leakage rate, which is not affected by $P_a$.

## 5.0 REGULATORY EVALUATION

### 5.1 Applicable Regulatory Requirements/Criteria

The proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met.

General Design Criterion 4, "Environmental and Dynamic Effects Design Bases," states that structures, systems and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

General Design Criterion 16, "Containment Design," states that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

General Design Criterion 19, "Control Room," states that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs, and that adequate radiation protection shall be provided. Adequate shielding is provided to maintain tolerable radiation levels in the control room under accident conditions for the duration of the accident.

General Design Criterion 38, "Containment Heat Removal," states that a system to remove heat from the reactor containment shall be provided that rapidly reduces, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptable low levels.

These General Design Criteria continue to be met by the change in $P_a$. The environmental qualification of equipment within containment is not affected by the change in $P_a$ following a LOCA. The change in $P_a$ will be reflected in future 10 CFR 50 Appendix J, Type A containment integrated leak rate testing, so containment integrity is not impacted by the change. The change in $P_a$ does not impact the maximum allowable containment leakage rate and therefore does not impact control room operator dose. $P_a$ remains below containment design pressure.

Based on the considerations above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) such activities will continue to be conducted in accordance with the site licensing basis; and (3) the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

In conclusion, EGC has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than the TS, and does not affect conformance with any regulatory requirements or criteria.

## 5.2 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests an amendment to Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed change would revise the peak calculated primary containment internal pressure for the design basis loss-of-coolant accident (LOCA) described in Technical Specifications (TS) 5.5.13, "Primary Containment Leakage Rate Testing Program." The peak calculated primary containment internal pressure, $P_a$, is increased from 39.9 psig to 42.6 psig.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change for LSCS, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

Criteria:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   **Response:** No.

   The proposed change to $P_a$ does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be increased by this proposed change since this change does not modify the plant or how it is operated.

   The change in $P_a$ will not affect radiological dose consequence analyses. LSCS radiological dose consequence analyses are based on the maximum allowable containment leakage rate. Even though the test pressure at which leak rate testing is performed is $P_a$, the maximum allowable containment leakage rate is defined in terms of a percentage of weight of the original content of containment air, which is independent of the peak calculated primary containment internal pressure. The Appendix J containment leak rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The consequences of an accident previously evaluated will not be increased by this proposed change.

   Therefore, operation of the facility in accordance with the proposed change to $P_a$ will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   **Response:** No.

   The proposed change provides a higher $P_a$ than currently described in the TS. This change is the result of a LOCA-Drywell Temperature sensitivity analysis performed by General Electric Hitachi (GEH). The peak calculated primary containment internal pressure remains below the containment design pressure of 45 psig. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   Therefore, operation of the facility in accordance with the proposed change to TS 5.5.13 would not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   **Response:** No.

   The peak calculated primary containment internal pressure remains below the containment design pressure of 45 psig. LSCS radiological consequence analyses are based on the maximum allowable containment leakage rate. The change in the peak calculated primary containment internal pressure does not represent a significant change in the margin of safety.

   Operation of the facility in accordance with the proposed change to TS 5.5.13 does not involve a significant reduction in the margin of safety.

Based on the above evaluation, EGC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

## 5.3 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by the operation of LSCS Units 1 and 2 in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

## 6.0 ENVIRONMENTAL CONSIDERATION

EGC has evaluated the proposed amendment for environmental considerations. The review has resulted in the determination that the proposed amendment would change requirements with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22, paragraph (b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

## 7.0 REFERENCES

1) GEH Report 0000-0149-2311-R0, Revision 0, "LaSalle County Generating Station Units 1 and 2, Short Term Containment Bounding Pa Assessment," dated August 2012

2) GE Project Task Report GE-NE-A1300384-02-01-R3, Revision 3, "LaSalle County Station Power Uprate Evaluation, Task 400: Containment System Response," dated May 2002 (documented in Design Analysis L-002874, Revision 0)

3) NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety," December 29, 1998

4) Operability Evaluation OE 11-002, Revision 5, "Drywell Temp Used as Input for the Containment Analysis," April 9, 2013

**ATTACHMENT 2**

**Markup of Technical Specifications Page**

**LASALLE COUNTY STATION**
**UNITS 1 AND 2**

**Docket Nos. 50-373 and 50-374**

**Facility Operating License Nos. NPF-11 and NPF-18**

REVISED TS PAGE

5.5-13

5.5 Programs and Manuals

5.5.13 Primary Containment Leakage Rate Testing Program (continued)

2. NEI 94-01 - 1995, Section 9.2.3: The first Unit 2 Type A test performed after December 8, 1993 Type A test shall be performed prior to startup following L2R12.

3. The potential valve atmospheric leakage paths that are not exposed to reverse direction test pressure shall be tested during the regularly scheduled Type A test. The program shall contain the list of the potential valve atmospheric leakage paths, leakage rate measurement method, and acceptance criteria. This exception shall be applicable only to valves that are not isolable from the primary containment free air space.

b. The peak calculated primary containment internal pressure for the design basis loss of coolant accident, $P_a$, is ~~39.9~~ 42.6 psig.

c. The maximum allowable primary containment leakage rate, $L_a$, at $P_a$, is 1.0% of primary containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Primary containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, the seal leakage rate is $\leq 5$ scf per hour when the gap between the door seals is pressurized to $\geq 10$ psig.

e. The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)