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

January 29, 2015

Mr. Bryan C. Hanson
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO)
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REVISING PEAK CALCULATED PRIMARY CONTAINMENT INTERNAL PRESSURE (TAC NOS. MF2690 AND MF2691)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 212 to Facility Operating License No. NPF-11 and Amendment No. 198 to Facility Operating License No. NPF-18 for the LaSalle County Station (LSCS), Units 1 and 2, respectively. The amendments are in response to your application dated September 5, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13249A231), as supplemented by letters dated June 12 and October 7, 2014 (ADAMS Accession Nos. ML14163A690 and ML14280A497, respectively).

The amendments increase the peak calculated primary containment internal pressure which is specified in LSCS Technical Specification (TS) 5.5.13, "Primary Containment Leakage Rate Testing Program." The amendments were requested to resolve a nonconservative TS.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions, please call me at 301-415-1380.

Sincerely,

Blake Purnell, Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 212 to NPF-11
2. Amendment No. 198 to NPF-18
3. Safety Evaluation

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 212
License No. NPF-11

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the LaSalle County Station, Unit 1 (the facility) Operating License No. NPF-11 filed by the Exelon Generation Company, LLC (the licensee) dated September 5, 2013, as supplemented by letters dated June 12 and October 7, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-11 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: January 29, 2015

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

EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-374

LASALLE COUNTY STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 198
License No. NPF-18

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the LaSalle County Station, Unit 2 (the facility) Operating License No. NPF-18 filed by the Exelon Generation Company, LLC (the licensee) dated September 5, 2013, as supplemented by letters dated June 12 and October 7, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-18 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 198, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Travis L. Tate, Chief
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications and Facility Operating License

Date of Issuance: January 29, 2015

ATTACHMENT TO LICENSE AMENDMENT NOS. 212 AND 198

FACILITY OPERATING LICENSE NOS. NPF-11 AND NPF-18

DOCKET NOS. 50-373 AND 50-374

Replace the following pages of the Facility Operating Licenses and Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License	License
NPF-11, Page 3	NPF-11, Page 3
NPF-18, Page 3	NPF-18, Page 3
TSs	TSs
5.5-13	5.5-13

Am. 146
01/12/01

(4) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

Am. 202
07/21/11

(5) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of LaSalle County Station, Units 1 and 2 and such Class B and Class C low-level radioactive waste as may be produced by the operation of Braidwood Station, Units 1 and 2, Byron Station, Units 1 and 2, and Clinton Power Station, Unit 1.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or thereafter in effect; and is subject to the additional conditions specified or incorporated below:

Am. 198
09/16/10

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3546 megawatts thermal).

Am. 212
1/29/15

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 212, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Am. 194
08/28/09

(3) DELETED

Am. 194
08/28/09

(4) DELETED

Am. 194
08/28/09

(5) DELETED

Am. 194
08/28/09

(6) DELETED

Am. 194
08/28/09

(7) DELETED

Am. 189
07/21/11

(5) Pursuant to the Act and 10 CFR Parts 30, 40, and 70 possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of LaSalle County Station, Units 1 and 2, and such Class B and Class C low-level radioactive waste as may be produced by the operation of Braidwood Station, Units 1 and 2, Byron Station, Units 1 and 2, and Clinton Power Station, Unit 1.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

Am. 185
09/16/10

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3546 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

Am. 198
1/29/15

(2) Technical Specification and Environment Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 198, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Am. 181
08/28/09

(3) DELETED

Am. 181
08/28/09

(4) DELETED

Am. 181
08/28/09

(5) DELETED

Am. 181
08/28/09

(6) DELETED

Am. 181
08/28/09

(7) DELETED

Am. 181
08/28/09

(8) DELETED

Am. 181
08/28/09

(9) DELETED

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 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)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 212 TO FACILITY OPERATING LICENSE NO. NPF-11

AND

AMENDMENT NO. 198 TO FACILITY OPERATING LICENSE NO. NPF-18

EXELON GENERATION COMPANY, LLC

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By application dated September 5, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13249A231), as supplemented by letters dated June 12 and October 7, 2014 (ADAMS Accession Nos. ML14163A690 and ML14280A497, respectively), Exelon Generation Company, LLC (the licensee) requested changes to the Technical Specifications (TSs) for the LaSalle County Station (LSCS), Units 1 and 2. The supplements to the application were provided in response to U.S. Nuclear Regulatory Commission (NRC) requests for additional information (RAIs) issued on April 10 and September 11, 2014 (ADAMS Accession Nos. ML14056A443 and ML14232A156, respectively). The supplemental responses provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on December 10, 2013 (78 FR 74182).

The proposed change would revise LSCS TS 5.5.13, "Primary Containment Leakage Rate Testing Program," by increasing the value of the peak calculated primary containment internal pressure (P_a) from 39.9 pounds per square inch gauge (psig) to 42.6 psig. The proposed change was requested to resolve a nonconservative TS. The application states that plant operations in TS 5.5.13 are being administratively controlled per NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications that are Insufficient to Assure Plant Safety."

2.0 REGULATORY EVALUATION

2.1 Description of the LSCS Containment Design

The LSCS primary containment design employs the boiling-water reactor Mark II concept of over-under pressure suppression with downcomers connecting the reactor drywell to the water filled suppression chamber (wetwell). The primary containment is a steel-lined, post-tensioned concrete enclosure which houses the reactor pressure vessel, the reactor coolant recirculation loops, and other principal reactor fluid connections such as the feedwater and main steam lines. A reinforced concrete floor, with penetrations for the downcomers, separates the drywell and wetwell. The primary containment provides one of the barriers to prevent the release of fission products to the environment. Isolation valves ensure that radioactive materials released from the reactor during postulated accidents remain within the primary containment and do not escape through containment penetrations. The containment structure is designed to withstand the peak transient pressures that could occur due to the postulated design-basis accident (DBA). The LSCS containment design pressure, as specified in updated final safety analysis report (UFSAR) Section 3.11.1.1.1, is 45 psig.

2.2 Description of the Proposed Change

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 application proposes to revise TS 5.5.13.b to state: "The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 42.6 psig."

The proposed change is to correct the nonconservative P_a value which the licensee identified as a result of a sensitivity analysis (revised analysis) performed by General Electric Hitachi (GEH) in 2012. The licensee's June 12, 2014, letter indicated that the revised analysis only considered the short-term containment response to a loss-of-coolant accident (LOCA) caused by the limiting event (UFSAR Section 6.2.1.1.3.1) of an instantaneous guillotine rupture of a recirculation line.

The application states that the analysis of record (AOR) for the LSCS containment used an initial drywell air temperature of 135 degrees Fahrenheit (°F), which is the maximum value allowed by TS 3.6.1.5, "Drywell Air Temperature." The application states that the revised analysis used a lower initial drywell air temperature of 98 °F and that this change resulted in a higher P_a than the AOR. In addition, the application states that the revised analysis corrects other secondary issues with the AOR, as follows:

- The AOR used the lower pressure relief mode setpoints for the main steam safety relief valves (SRVs). The revised analysis used the higher pressure safety mode setpoints for the main steam SRVs as specified in TS 3.4.4, "Safety/Relief Valves (S/RVs)."
- The AOR used a main steam isolation valve (MSIV) closure time of 5 seconds and the revised analysis used an MSIV closure time of 3.5 seconds.
- The AOR used a reactor power level of 3789 MegaWatts thermal (MWt)[1] and the revised analysis used a power level of 3559 MWt.

[1] The application states that the AOR incorrectly stated it was performed at 3559 MWt.

- The revised analysis considered different core flow and feedwater temperature conditions, and the results are based on the conditions that yield the highest P_a.

2.3 Regulatory Review

The following NRC regulatory requirements were applied during the NRC staff's review of the proposed amendments.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants," establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. Section 1.2.1 of the LSCS UFSAR provides a discussion of the principal design criteria and the safety design criteria that were applied in the design, fabrication, and erection of the plant. Section 3.1, "Conformance with NRC General Design Criteria," of the UFSAR evaluates the LSCS design basis against the GDC and concludes that the LSCS fully satisfies and is in compliance with the GDC.

The NRC staff acceptance criteria for the primary containment functional design are based on the following GDC:

- GDC 4, "Environmental and Dynamic Effects Design Bases," insofar as it requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and that SSCs be appropriately protected against dynamic effects.

- GDC 16, "Containment Design," insofar as it requires that the containment and associated systems be designed 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 the postulated accident requires.

- GDC 19, "Control Room," insofar as it requires that actions can be taken from the control room to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs, and with adequate radiation protection to maintain prescribed radiation levels in the control room under accident conditions for the duration of the accident.

- GDC 38, "Containment Heat Removal," insofar as it requires that a containment heat removal system be provided and that its function shall be to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

- GDC 50, "Containment Design Basis," insofar as it requires that the containment structure and its associated heat removal systems be designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Paragraph 50.36(c)(2)(ii) of 10 CFR 50.36, "Technical Specifications," provides criteria for establishing TS limiting conditions for operation. Criterion 2 requires a TS limiting condition for operation for a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3.0 TECHNICAL EVALUATION

3.1 Short-Term Containment Response Analysis

The licensee's June 12, 2014, letter states that the AOR assessed both the short-term and long-term containment pressure and temperature response. The letter states that the short-term analysis determines the peak containment (drywell) pressure. Therefore, as stated above, the revised analysis only considered the short-term containment response to a LOCA caused by the limiting event of an instantaneous guillotine rupture of a recirculation line.

3.1.1 Methodology

The licensee's June 12, 2014, letter states that the models and methodology, which are comprised of several computer codes, used to determine P_a for the short-term response of the AOR and the revised analysis are the same. The computer code used to calculate the mass and energy released from the break is LAMB-08 for both analyses. The revised analysis used different LAMB code options than the AOR for the critical break flow and the core heat flux based on the current GEH containment analysis methodology. The computer codes used to determine the short-term pressure and temperature responses of the containment (drywell) are M3CPT05V for the AOR and M3CPT05 for the revised analysis; these codes are the same but run on different environments. The models and methodology used to determine P_a in the revised analysis are acceptable to the NRC staff because the licensee's previously approved AOR used the same models and methodology for calculating the short-term containment response at the LSCS.

3.1.2 Initial Drywell Temperature

The application states that to determine P_a the revised analysis used an initial drywell air temperature of 98 °F, where the AOR used 135 °F. The application states:

> The 2012 GEH Analysis uses an initial drywell temperature of 98 °F based upon historical bulk average drywell temperature [BADT] 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 [percent] power level, the drywell temperature will be ≥ 100 °F. The drywell temperature of 98 °F was selected to add conservatism.

The licensee's October 7, 2014, letter describes its evaluation of BADT data for the LSCS covering the past three refueling outages (February 2010 to March 2014 for Unit 1 and January 2009 to March 2013 for Unit 2). These data were also taken in 1-hour increments and covered

operations in Modes 1, 2, and 3, which are the modes where the primary containment is required to be operable. Separate histograms were provided for each unit showing the temperature distribution of the BADT data for all Mode 1 operations. The letter also provides graphs covering Modes 1, 2, and 3 for each unit showing the power level associated with each BADT measurement less than 100 °F.

The licensee's October 7, 2014, letter states that the revised analysis included parametric studies evaluating the impact of reactor power level on P_a. The letter includes the results for a parametric study using a constant initial drywell air temperature of 98 °F and 100 percent rated core flow. The study considered five power levels ranging from 2163 MWt (61 percent) to 3559 MWt (100.4 percent). The results showed that P_a increased from 40.7 psig at the lowest power lever to 42.6 psig at the highest power level (DBA case).

The October 7, 2014, letter also states that there was a linear relationship between P_a and the initial drywell air temperature, such that a lower initial temperature would result in a higher P_a. The licensee used this relationship with the results of the parametric study to determine P_a for various initial drywell air temperatures and reactor power levels. The letter provides the results of this determination in a graph showing P_a versus initial drywell air temperature (ranging from 60 °F to 98 °F) for the five reactor power levels considered in the parametric study. Based on this information, the licensee states that for P_a to exceed 42.6 psig with an initial drywell air temperature of 80 °F the reactor power level must be above approximately 91 percent (approximately 3226 MWt). The graph shows that P_a will not exceed 42.6 psig if the initial drywell air temperature is 60 °F or higher and the power level is 2861 MWt (80.7 percent) or less.

The NRC staff evaluated the information on the BADT data using the relationship between initial drywell air temperature and reactor power on P_a as represented on the graph provided by the licensee. For Modes 2 and 3, the October 7, 2014, letter states that the highest power level for the two units was 8 percent and the lowest BADT observed was 79 °F. Based on this, the staff determined that a P_a of 42.6 psig bounds operations in Modes 2 and 3 since the power levels will be significantly below 80.7 percent and the lowest BADT observed is significantly above 60 °F.

The histograms show that the BADT for both units is typically much greater than 98 °F for Mode 1 operation, with a BADT above 108 °F for Unit 1 and above 102 °F for Unit 2 for a significantly high number of data sets. For the period of data evaluated, the October 7, 2014, letter indicates that during Mode 1 operations, for each unit, the power level did not exceed 40 percent when the BADT was 100 °F or less and the power level did not exceed 25 percent when the BADT was 98 °F or less. The letter also states that the highest power level associated with a BADT of 100 °F during the period the data was taken was 725 MWt (20 percent) for Unit 1 and 1355 MWt (38 percent) for Unit 2. In addition, the application states that the revised analysis included a statistical analysis of the BADT for a different time period and determined that when the plant operates at or above 90 percent power level the BADT will be at least 100 °F. Based on this, the staff determined that the BADT is likely to exceed 98 °F for power levels at or above 90 percent. The NRC staff also determined that for a BADT of less than 98 °F the power level will likely be significantly lower than what would be required for P_a to exceed 42.6 psig based on the licensee's analysis described in the October 7, 2014, letter.

As discussed above, the NRC staff reviewed the licensee's evaluation of data regarding the relationship between power level and BADT. The staff compared this evaluated data with the licensee's analysis showing how P_a varies with power level and initial drywell air temperature. Based on this, the staff determined that for both units at the LSCS a P_a of 42.6 psig adequately bounds the full range of likely power operations with respect to power level and BADT.

3.1.3 Other Input Parameters

The application states that the LSCS main steam SRVs are designed to open in both safety and relief modes of operation. The AOR used the lower pressure relief mode setpoints for the SRVs, and the revised analysis used the higher pressure safety mode setpoints which are specified in TS 3.4.4. The application states that the higher setpoints would direct more energy through the break to the drywell.

The LSCS TS Surveillance Requirement 3.6.1.3.6 specifies that the closure time for the MSIVs shall be between 3 and 5 seconds. The application states that the AOR used the maximum allowed MSIV closure time of 5 seconds. The revised analysis used a shorter MSIV closure time of 3.5 seconds, based on the minimum MSIV closure time of 3 seconds with a 0.5 second delay between break occurrence and MSIV initiation. The application states that a shorter MSIV closure time results in a higher P_a since more reactor coolant mass and energy would be directed to the drywell through the break.

The application states that the revised analysis considered different core flow and feedwater temperature conditions, and that its results are based upon the conditions that yield the highest P_a. Since core flow conditions used in the analysis affect the mass and energy release through the break, GEH evaluated plant operation at rated core flow, increased core flow, and maximum extended load limit analysis conditions to ensure that the condition which maximizes the energy release to the drywell is accounted for. Similarly, final feedwater temperature reduction and normal feedwater temperature were considered and the condition that bounds P_a was used in the revised analysis.

The application states that the AOR was performed at a reactor power level of 3789 MWt. The revised analysis used a maximum power level of 3559 MWt, which is still higher than the licensed power level of 3546 MWt. A lower reactor power level results in a lower P_a since less energy would be released through the break to the drywell.

Based on the information above, the NRC staff determined that input changes related to SRV setpoints, MSIV closure time, core flow conditions, and feedwater temperature conditions are conservative, as these changes result in a higher P_a than the AOR. Therefore, these input changes are acceptable to the staff. The change to reactor power level input is acceptable because it bounds the licensed power level with sufficient margin to account for uncertainties due to instrument error (ADAMS Accession No. ML101830361).

3.1.4 Conclusions Regarding the Revised Analysis

The results of the revised analysis show that P_a has increased from 39.9 psig to 42.6 psig. Based on the discussion above, the NRC staff finds that the proposed P_a has been determined

using acceptable methods because its methods were the same as those that had been used in the licensee's previously approved AOR. Additionally, the NRC staff finds that the proposed P_a has been determined using acceptable input parameters because each input parameter is appropriately conservative. Since the proposed P_a was determined using acceptable methods and input parameters and since it remains below the LSCS containment design pressure of 45 psig, the NRC staff concludes that it is acceptable.

3.2 Operating Restrictions

The current containment DBA analysis (AOR) for the LSCS is described in UFSAR Section 6.2.1.1.3. Initial conditions for this analysis include the maximum drywell and suppression chamber pressure and the maximum drywell air temperature. According to the LSCS TS bases (ADAMS Accession No. ML14113A133), these initial conditions satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) and, therefore, are included as limiting conditions for operation in TS 3.6.1.4, "Drywell and Suppression Chamber Pressure," and TS 3.6.1.5, "Drywell Air Temperature," respectively. As part of its review, the NRC staff considered whether additional operating restrictions were needed since the lower initial drywell air temperature used in the revised analysis of 98 °F is not included in the LSCS TSs. Instead, limiting conditions for operation 3.6.1.5 only states that, "Drywell average air temperature shall be ≤ 135 °F."

The licensee has proposed a P_a of 42.6 psig which was determined using an initial drywell air temperature of 98 °F and a power level of 3559 MWt (100.4 percent). In Section 3.1.2, the NRC staff considered the relationship between P_a, power level, and drywell air temperature. For both units at the LSCS, the staff determined that a P_a of 42.6 psig adequately bounds the full range of likely power operations with respect to power level and BADT. In Section 3.1.4, the staff found the proposed P_a to be acceptable because it remains below the LSCS containment design pressure of 45 psig.

The licensee's October 7, 2014, letter provides the linear equation relating P_a to the initial drywell air temperature. Using this equation, the NRC staff determined that the initial drywell air temperature would have to be less than 30 °F before P_a exceeds 45 psig at a power level of 3559 MWt. This temperature is significantly below any of the BADT data provided with the letter. Based on this, the staff determined that there is significant margin between P_a and the containment design basis pressure.

Based on the above, the NRC staff concludes that additional operating restrictions are not needed to ensure that the plant operates within the bounds of the revised analysis.

3.3 Containment Leak Rate Testing

The application states that the LSCS containment leak rate testing program is in accordance with Option B, "Performance-Based Requirements," Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50 as modified by approved exemptions and criteria contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program" (ADAMS Accession No. ML003740058). At LSCS, the radiological dose consequences of a design basis LOCA are based, in part, on the maximum allowable primary containment leakage rate (L_a). Currently, TS 5.5.13 specifies that L_a is

1 percent of primary containment air weight per day and that leak testing is based on a P_a of 39.9 psig. The proposed higher P_a of 42.6 psig will increase the measured leakage.

The licensee's June 12, 2014, letter states that the last two integrated leak rate tests (ILRTs) were performed in 1994 and 2008 for Unit 1, and 1993 and 2009 for Unit 2. The minimum test pressures for all these tests were higher than 39.9 psig but less than 42.6 psig. The results from the most recent ILRTs measured a leakage rate of 0.472 percent and 0.386 percent of primary containment air weight per day for Units 1 and 2, respectively. The letter states that by modeling the leakage as compressible flow through an orifice the most recent ILRT results would be expected to increase by 0.14 percent for Unit 1 and 1.68 percent for Unit 2 if they were performed at 42.6 psig. This equates to an expected leakage rate of approximately 0.473 percent and 0.392 percent of primary containment air weight per day for Units 1 and 2, respectively. Based on this, the NRC staff determined that there will be significant margin between the expected leakage rate and L_a.

The June 12 and October 7, 2014, letters provide the local leak rate test (LLRT) results (Type B and C) for the 2012 Unit 1 and 2013 Unit 2 refueling outages. In accordance with the LSCS Appendix J testing program, only a portion of the primary containment isolation valves are tested in any one outage. The minimum test pressures reported in these letters were higher than 39.9 psig. The combined (Type B and C) as-left minimum and maximum pathway leakages are 101.23 standard cubic feet per hour (scfh) and 206.39 scfh for Unit 1, and 73.38 scfh and 160.93 scfh for Unit 2. The as-found minimum pathway leakages are 104.32 scfh for Unit 1 and 110.37 scfh for Unit 2. The licensee compared these results to the allowable leakage rate of 364.4 scfh and determined that there is a margin varying from 43 to 80 percent.

Based on a review of the information provided by the licensee, the NRC staff determined that increasing P_a from 39.9 psig to 42.6 psig would not have a significant impact on future containment leak rate test results and that significant margin will be maintained between the expected leakage rates and the allowed leakage rates.

3.4 Technical Conclusion

The NRC staff concludes that the proposed change meets GDC 4 and GDC 16 because the licensee showed that the pressure in the LSCS containment under design basis LOCA conditions does not exceed the maximum containment design pressure of 45 psig. The staff concludes that the proposed change meets GDC 19 because the licensee has shown that considerable margin exists between the expected and allowable leakage rates for primary containment and the primary containment isolation valves. The staff concludes that GDC 38 will continue to be met because the proposed change in short-term pressurization will have an insignificant impact on long-term cooling. The staff concludes that the proposed change meets GDC 50 because the proposed P_a is less than the containment design-basis pressure and because there is adequate margin between the expected leakage rates and the containment design-basis leakage rate. For these reasons, the proposed change is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding as published in the *Federal Register* on December 10, 2013 (78 FR 74182). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributors: Nageswara Karipineni, NRR/SCVB
Blake Purnell, NRR/DORL

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If you have any questions, please call me at 301-415-1380.

Sincerely,

/RA/

Blake Purnell, Project Manager
Plant Licensing III-2 and
Planning and Analysis Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 212 to NPF-11
2. Amendment No. 198 to NPF-18
3. Safety Evaluation

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