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

February 14, 2019

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS TO RENEWED FACILITY OPERATING LICENSES RE: REVISION OF TECHNICAL SPECIFICATIONS TO ADOPT TSTF-334, "RELAXED SURVEILLANCE FREQUENCY FOR EXCESS FLOW CHECK VALVE TESTING RELATED TO EXCESS FLOW CHECK VALVE TESTING" (EPID L-2018-LLA-0035)

Dear Mr. Hanson:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment No. 235 to Renewed Facility Operating License No. NPF-11 and Amendment No. 221 to Renewed Facility Operating License No. NPF-18 for the LaSalle County Station (LSCS), Units 1 and 2, respectively. The amendments revise the relevant portions of the technical specification (TS) and license pages in response to your application dated February 7, 2018.

The amendment implements Technical Specification Task Force (TSTF)-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000 by revising surveillance requirement (SR).

The amendments revise SR 3.6.1.3.8 in LSCS TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," to verify that a representative sample of reactor instrumentation line excess flow check valves (EFCV) are tested, in accordance with the Surveillance Frequency Control Program (SFCP), such that each EFCV will be tested at least once every 10 years (nominal). The LSCS SFCP currently requires each EFCV to be tested on a 24-month frequency for SR 3.6.1.3.8.

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

Sincerely,

Bhalchandra K. Vaidya, Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosures:
1. Amendment No. 235 to NPF-11
2. Amendment No. 221 to NPF-18
3. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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EXELON GENERATION COMPANY, LLC

DOCKET NO. 50-373

LASALLE COUNTY STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 235
Renewed License No. NPF-11

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated February 7, 2018, 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 Renewed Facility Operating License No. NPF-11 is hereby amended to read as follows:

    (2) <u>Technical Specifications and Environmental Protection Plan</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 235, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed 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 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Renewed Facility
Operating License No. NPF-11 and Technical
Specification Pages

Date of Issuance: February 14, 2019

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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 221
Renewed License No. NPF-18

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

   A. The application for amendment filed by the Exelon Generation Company, LLC (the licensee), dated February 7, 2018, 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 Renewed 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. 221, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed 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 45 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Renewed Facility
Operating License No. NPF-18 and Technical
Specification Pages

Date of Issuance: February 14, 2019

ATTACHMENT TO LICENSE AMENDMENT NOS. 235 AND 221

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

LASALLE COUNTY STATION, UNITS 1 AND 2

DOCKET NOS. 50-373 AND 50-374

Replace the following pages of the Renewed Facility Operating Licenses and Appendix A, Technical Specifications (TSs), with the attached revised 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-18, Page 3	NPF-11, Page 3 NPF-18, Page 3
TSs	TSs
TS 3.6.1.3-8	TS 3.6.1.3-8

(3) Exelon Generation Company, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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 renewed 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. 198
09/16/10

(1) <u>Maximum Power Level</u>

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

Am. 235
02/14/19

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 235, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed 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

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) 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. 189
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 renewed 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. 221
02/14/19

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 221, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.8	Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.3.10	Verify leakage rate through any one main steam line is ≤ 200 scfh and through all four main steam lines is ≤ 400 scfh when tested at ≥ 25.0 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.11	Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 235 TO RENEWED FACILITY OPERATING

LICENSE NO. NPF-11 AND AMENDMENT NO. 221 TO RENEWED 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 February 7, 2018, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18039A123), Exelon Generation Company, LLC (EGC or the licensee), requested amendments to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for the LaSalle County Station (LSCS), Units 1 and 2, respectively. In addition, the application requested approval of an associated relief request, which was granted by the NRC by letter dated July 3, 2018 (ADAMS Accession No. ML18163A054).

The proposed amendments would adopt the Technical Specification Task Force (TSTF) Traveler TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valves [EFCV] Testing" approved by the NRC staff on October 31, 2000, by letter from W. D. Beckner to A. R. Pietrangelo, Nuclear Energy Institute (Reference 3). TSTF-334 proposed specific changes to the Standard Technical Specifications (STS) providing guidance for licensees implementing the extended excess flow check valves (EFCV) surveillance test intervals. Adoption of TSTF-334 is only permitted for those plants for which Boiling water Reactor Owners Group (BWROG) Topical Report (TR) NEDO-32977-A "Excess Flow Check Valve Testing Relaxation" (Reference 1) is applicable and which are subject to EFCV performance and corrective action criteria. The licensee affirmed the applicability of TSTF-334, Revision 2, in its application dated February 7, 2018, with specific deviations. Approval of this request would result in each EFCV being tested at least once every 10 years (nominal).

TSTF-334 Revision 2 (Reference 3) provides a revision to plant-specific surveillance requirement (SR) equivalent to Standard Technical Specifications (STS) 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," (NUREG-1433), to allow a representative sample (i.e., approximately 20 percent) of reactor instrumentation line excess flow check valves to be periodically tested during each refueling outage in accordance with the plant surveillance frequency control program (SFCP). Prior to the development of TSTF-334, typical BWR plant technical specifications (TSs) required testing of each EFCV once every refueling outage. The industry-identified need for this TSTF was based on a recognition that there are a large number

of EFCVs at BWR plants (around 100 per unit) that are required to be tested each refueling outage and a "significant cost and dose savings can be achieved by the proposed relaxation of the testing frequency without any reduction in overall safety or reliability." The justification supporting TSTF-334 was that "operating experience demonstrates that EFCVs are highly reliable and that the incidence of test failures is extremely low." The TSTF includes a commitment for licensees to evaluate any failure of a valve to isolate during performance testing, and to determine if additional testing (i.e., increased sample size) within that test interval is warranted to ensure the overall reliability (of valve actuation on simulated demand) maintained.

2.0 REGULATORY EVALUATION

2.1 System Description

The EFCVs in reactor instrumentation lines are used in boiling-water reactor (BWR) containments to limit the release of fluid from the reactor coolant system in the event of an instrument line break. EFCVs are not required to close in response to a containment isolation signal and are not postulated to operate under post-loss-of-coolant accident (LOCA) conditions. As noted in TR NEDO-32977-A (Reference 1), EFCVs are not needed to mitigate the consequences of an accident because an instrument line break coincident with a design basis LOCA would be of a sufficiently low probability to be outside of the design basis.

EFCVs are used in the instrument sensing lines of reactor protection systems, engineered safeguards systems, and reactor control systems of BWRs for sensing lines that are either (a) directly connected to the reactor coolant pressure boundary (RCPB), (b) open to the containment, or (c) connected to closed piping systems within the containment. The EFCVs limit the release of fluid from the reactor coolant system or contaminated air in the containment in the event of an instrument line break or gross leakage from an instrument line downstream of the EFCV. The manufacturer of the LSCS Unit 1 and 2 EFCVs is Dragon Valves. The licenses test data show the LSCS EFCVs are similar to the valves used by the other member utilities in Reference 1. Additionally, a failure rate analysis of the LSCS EFCVs was performed over six operating cycles.

Current TS SR 3.6.1.3.8 requires the licensee to "[v]erify each reactor instrumentation line EFCV actuates to the isolation position on an actual or simulated line break signal" at a frequency "[i]n accordance with the Surveillance Frequency Control Program [SFCP]." The current frequency provided in the SFCP is once per refueling outage.

The LSCS UFSAR, Section 15.6.2, "Instrument Line Break," identifies that instrument lines penetrating containment at LSCS from the RCPB emergency core cooling system (ECCS) are designed to high quality engineering codes and standards and seismic and environmental requirements. The lines are equipped with a 1/4-inch diameter flow-restricting orifice inside the primary containment. Due to the slow rate of coolant loss, an orderly reactor system depressurization follows reactor scram, and the primary system is cooled down and maintained without ECCS activation. No fuel damage or core uncover occurs as a result of an instrument line break.

2.2 Description of the Proposed Changes

The license amendment request (LAR) proposes adoption of TSTF-334, Revision 2 (Reference 3), to revise the LSCS TS SR 3.6.1.3.8 and its associated TS Bases. The proposed amendment would replace the requirement to verify the operability of each instrument sensing line EFCVs during each refueling outage with a requirement to verify the operability of a sample of approximately 20 percent of the EFCV population. This would result in each EFCV being tested approximately once every 10 years assuming a 24-month refueling outage cycle. The licensee stated that the basis for increasing the surveillance interval is to minimize personnel exposure associated with the performance of periodic inservice performance testing of the functionality of the EFCVs, and to reduce the outage time currently allocated to accomplish such testing. In the LAR, the licensee argued that recent improvements in refueling outage schedules minimize the time that is planned for refueling and testing activities during the outage. As a result of the shortened outages, decay heat levels during hydrostatic tests that must be performed to support EFCV testing are higher than in past outages. The LAR states: "[i]f such hydrostatic conditions were extended to accommodate testing of all installed plant EFCVs, the reactor vessel could require several depressurizations to avoid exceeding the maximum bulk coolant temperature limit. This evolution challenges the reactor operators and thermally cycles the reactor vessel, and therefore should be avoided if possible."

The current LSCS TS SR 3.6.1.3.8 is as follows:

> Verify each reactor instrumentation line EFCV actuates to the isolation position on an actual or simulated instrument line break signal.

In the LAR, the licensee proposes to change TS SR 3.6.1.3.8 by removing the word "each" and replacing it with "a representative sample." The revised SR would state:

> Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.

The proposed amendment does not change the frequency for SR 3.6.1.3.8 which continues to state "[i]n accordance with Surveillance Frequency Control Program [SFCP]".

The LAR also contained relief request (RR) RV-02 Related to EFCV Testing Frequency," which was submitted in accordance with 10 CFR 50.55a, "Codes and standards." RR RV-02 requested relief from portions of the inservice testing program requirements of the American Society of Mechanical Engineers Operation and Maintenance (ASME OM) Codes ISTC-3522(a), ISTC-3522(c), and ISTC-3700. The NRC granted the licensee's request for relief from this code requirement in its letter of July 3, 2018 (ADAMS Accession No. ML18163A054). The staff's decision to grant relief was based on the demonstrated reliability of the LSCS EFCVs to close when periodically tested, and on the staff's finding that the maintenance history over six 24-month operating cycles from February 2005 through February 2017 revealed a failure rate consistent with the findings in TR NEDO-32977-A (Reference 1).

2.3 Regulatory Requirements and Guidance Used in the Evaluation of the Proposed Change

The NRC staff identified the following regulatory requirements and guidance as being applicable to the proposed amendment:

2.3.1 Regulatory Requirements

- 10 CFR Section 50.36, "Technical specifications," provides the regulatory requirements for the content of the TSs. Specifically, 10 CFR 50.36(c) requires that TSs include items in five specific categories related to station operation. These categories are: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting condition for operations (LCOs); (3) SRs; (4) design features; and (5) administrative controls.

- 10 CFR 50.36(c)(3), "Surveillance requirements," states that "[s]urveillance requirements are requirements related to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

- 10 CFR 50.55a(b) requires, in part, that systems and components of boiling water power reactors to meet the requirements of the ASME OM Code.

- 10 CFR 50.55a(h) "Protection and safety systems," provides the requirements for power reactors protection systems Specifically, 10 CFR 50.55a(h)(2) provides:

> (2) *Protection systems*. For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements in IEEE [Institute of Electrical and Electronics Engineers] Std [Standard] 279-168, "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems," or the requirements in IEEE Std 279–1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," or the requirements in IEEE Std 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dates January 30, 1995.

- IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," Section 1 states that protection system signals "include those that actuate reactor trip and that, in the event of a serious reactor accident, actuate engineered safeguards such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning." Section 4.20 further states "[t]he protection system shall be designed to provide the operator with accurate, complete, and timely information pertinent to its own status and to generating station safety. The design shall minimize the development of conditions which would cause meters, annunciators, recorders, alarms, etc. to give anomalous indications confusing to the operator."

- 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants" states, in part:

> Each holder of an operating license for a nuclear power plant under this part . . . shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components, as defined in paragraph (b) of this section, are capable of fulfilling their intended functions. These goals shall be established

commensurate with safety and, where practical, take into account industrywide operating experience.

- 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," provides the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

  - o 10 CFR 50, Appendix A, GDC 13, "Instrumentation and control," states:

    Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

  - o 10 CFR 50 Appendix A, GDC 21, "Protection system reliability and testability," states, in part:

    The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed . . . The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

  - o 10 CFR 50, Appendix A, GDC 55, "Reactor coolant pressure boundary penetrating containment," states, in part:

    Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis.

  - o 10 CFR 50, Appendix A, GDC 56, "Primary containment isolation," states, in part:

    Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis.

- 10 CFR 100.11(a) requires a license applicant to determine:

    (1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem [roentgen equivalent man, a unit used to measure the does equivalent, which is a measure of the biological damage to living tissue as a result of radiation exposure] or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

    (2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.9.

2.3.2 Guidance

- NUREG-0800, Standard Review Plan (SRP), Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2 (Reference 4), provides guidance to the NRC staff for the review of the radiological consequences of failures outside the containment of small lines connected to the primary coolant pressure boundary, such as instrument lines and sample lines. SRP Section 15.6.2 states, in part, that the NRC reviewer should evaluate the proposed change against 10 CFR 100.11, such that the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed a small fraction of the exposure guideline values of 10 CFR 100.11. A "small fraction" of 10 CFR Part 100 means 10 percent of these exposure guideline values, that is, 2.5 rem and 30 rem for the whole-body and thyroid doses, respectively.

- Regulatory Guide (RG) 1.11, Revision 1, "Instrument Lines Penetrating the Primary Reactor Containment" (Reference 5) describes a suitable basis which may be used to implement GDC 55 and 56 for demonstrating the acceptability of instrument sensing lines.

    - RG 1.11 provides important distinctions in design guidance for instrument lines that serve protection system instrument functions from instrument lines that do not serve protection functions. Specifically, RG 1.11 states that:

        > Lines connected to instruments that are part of the protection system are extensions of that system and should satisfy the requirements for redundancy, independence, and testability for the protection system, to assure that the protective function will be accomplished. Lines connected only to instruments that are not part of the protection system need not meet the requirements of the protection system. For these [non-protection system] lines, the assurance that isolation can be effected when required is of

greater importance to safety than the capability of the connected instrument function.

This guidance indicates that for instrument lines that serve instruments required to accomplish protection functions, it is of primary importance for the design of instrument lines penetrating the primary containment to assure the reliable accomplishment of protective functions, while the line's isolation capabilities are secondary to the need for assuring the reliability of protective functions.[1]

- o RG 1.11 states "[i]t is desirable to have an indication of the valve position status (opened and closed) in the control room because without such an indication, a valve may remain closed, thus impairing instrument functions for an excessive period of time."

- NUREG-1433, Revision 4, Vol. 1, "STS - GE Electrical Plant (BWR/4)" (Reference 6) STS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)" provides the surveillance requirement and frequency for testing each reactor instrumentation line EFCV. Specifically, STS SR 3.6.1.3.10 currently requires verification of the actuation capability of either each reactor instrumentation line EFCV or a representative sample of reactor instrumentation line EFCVs every 18 months or in accordance with the SFCP.

- The NRC safety evaluation (SE), dated March 14, 2000, associated with the approval of TR NEDO 32977-A (Reference 1) states the NRC's position on the industry's implementation plan for relaxing the EFCV surveillance frequency. The NRC agreed that the test interval could be extended up to a maximum of 10 years. In conjunction with this finding, the NRC noted that each licensee that adopts the relaxed test interval program for EFCVs must have a failure feedback mechanism and corrective action program to ensure that EFCV performance and reliability continues to be bounded by the TR NEDO 32977-A results. Each licensee was required to perform a plant-specific radiological dose assessment, EFCV failure analysis, and release frequency analysis to confirm that their results are bounded by the generic analyses of the TR NEDO 32977-A.

2.3.3 LSCS Updated Final Safety Analysis Report (UFSAR)

LSCS UFSAR (Reference 7) Appendix B, "Conformance to Regulatory Guides" discusses conformance to RG 1.11 and states:

> Regulatory Guide 1.11 describes an acceptable basis on which to demonstrate the acceptability of instrument lines which penetrate or form a part of the reactor primary containment. Instrument sensing lines that penetrate or connect to the primary reactor containment are designed to strict penetration requirements as discussed in Subsection 6.2.4 and shown in Table 6.2-21. The provisions of this regulatory guide are incorporated into the design of the instrument sensing lines that penetrate or connect the primary containment at LSCS. We believe that we comply with the guidance set forth in this guide.

[1] Per Section 1 of IEEE Standard 279-1971, protection system signals "include those that actuate reactor trip and that, in the event of a serious reactor accident, actuate engineered safeguards such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning."

3.0 TECHNICAL EVALUATION

LSCS submitted an LAR and a RR to extend the testing frequency for EFCVs by implementing TSTF-334, Revision 2 (Reference 3). LSCS SR 3.6.1.3.8 currently states "[v]erify each reactor instrumentation line EFCV actuates to the isolation position on an actual or simulated instrument line break signal" at a frequency "in accordance with the Surveillance Frequency Control Program [SFCP]." The SR determines that each EFCV is operable by confirming that the valve actuates to check flow on a simulated instrument line break downstream of the valve. The LSCS Units 1 and 2 SFCPs currently require each EFCV to be tested on a 24-month frequency.

According to TSTF-334, Revision 2 (Reference 3), to receive approval to adopt TSTF-334, the licensee must provide at least the following two sets of information: (a) demonstrate that the EFCV performance and failure rate for their plant EFCVs is comparable to that which has been documented in TR NEDO-32977-A (Reference 1); and (b) provide the EFCV performance test criteria and basis for NRC staff review and approval.

3.1 EFCV Failure Rate

To determine whether LSCS's EFCV performance and failure rate is comparable to the failure rates documented in TR NEDO-32977-A (Reference 1), the NRC staff reviewed LSCS EFCV failure rate analysis and release frequency. The performance and failure rate for LSCS's EFCVs, which are manufactured by Dragon Valves, was evaluated in GE Nuclear Hitachi Nuclear Energy Report 004N6095, Revision 0, "LaSalle County Station Excess Flow Check Valve (EFCV) Failure Rate Analysis" (Reference 2) and was provided as Attachment 2 in the licensee's LAR. The LSCS EFCV failure rate analysis was conducted over six 24-month operating cycles from February 2005 through February 2017 and identified 10 EFCVs failures attributed to the valve failing to check flow.

The NRC staff reviewed the details regarding each of the EFCV failures at LSCS, which are provided in GE Report 004N6095, Table 3-1 (Reference 2). The staff compared the LSCS EFCV failure rates to those listed in TR NEDO-32977-A (Reference 1). Specifically, TR NEDO-32977-A, Table 4-2 lists the EFCV failure rates by manufacturer. Table 4-2 shows that EFCVs manufactured by Dragon Valves were determined to have an upper limit failure rate of 2.89E-7 per hour. The analysis of LSCS plant-specific EFCV failure rate provided in GE Report 004N6095, Table 3-2 indicates that the calculated upper limit of expected failures is 1.14E-6 and 8.80E-7 per hour for the EFCV in Unit 1 and 2, respectively. The LSCS EFCV failure rate analysis data is consistent with the Dragon EFCV failure rate in TR NEDO-32977-A and is generally consistent with the EFCV failure rate analysis for all valve manufacturers at the twelve BWR plants evaluated in TR NEDO-32977-A (Reference 1).

Based on the above discussion the NRC staff finds that failure rate data of LSCS EFCVs provided in GE Report 004N6095 (Reference 2) is comparable with TR NEDO-32977-A (Reference 1). This demonstrates that TSTF-334 is applicable to LSCS Units 1 and 2. As such, the staff concludes that extending the testing frequency of the EFCVs will assure that the EFCVs are maintained, that facility operation will be within the Safety Limits, and that the LCO will continue to be met.

3.2 Licensee's Failure Feedback Mechanism and Corrective Action Program

The NRC staff notes that TR NEDO-32977-A (Reference 1) does not provide a specific failure feedback mechanism, but does state that a plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions for TSTF-334 to be applicable. In responding to an NRC staff request for addition information on TR NEDO-32977-A concerning the failure feedback mechanism, the BWROG stated that each licensee that adopts the relaxed surveillance intervals recommended by the TR should ensure that an appropriate feedback mechanism responsive to EFCV failure trends is in place.

In the LAR, the licensee states that as part of the proposed changes the LSCS Maintenance Rule program would be revised to provide a means to track the performance of the EFCVs in a manner similar to existing performance-based testing programs, such as Option B to 10 CFR Part 50, Appendix J. The licensee notes that to ensure that the EFCV performance remains consistent with the extended test interval, a minimum performance standard has been established. The performance standard provided by the licensee in the LAR requires less than or equal to two functional failures during a 24-month rolling average to ensure that adverse trends in EFCV performance are identified and dispositioned in the corrective action program.

The NRC staff notes that 10 CFR, Part 50, Appendix J, testing is only applicable to EFCVs if they perform a containment isolation function. A containment isolation valve (CIV) is defined in Appendix J as "any valve which is relied upon to perform a containment isolation function." Appendix J prescribes air testing requirements for containment isolation valves and provides for exemptions for valves which are water sealed. Most EFCVs are connected to water-filled systems, and are tested for operability with water. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post-LOCA conditions. Consequently, for purpose of 10 CFR 50, Appendix J, CIV testing, EFCVs do not provide a containment isolation function.

Nevertheless, the NRC staff has determined that the proposed changes to the LSCS Maintenance Rule program that track the performance of the EFCVs conform to Option B "Performance-Based Requirements" of Appendix J to 10 CFR Part 50, which identifies a risk-informed, performance based approach to leakage rate testing of containment isolation valves. Because the proposed changes are similar to the method approved in Option B of Appendix J to 10 CFR Part 50, the NRC staff finds that the proposed failure feedback mechanism and corrective action program are acceptable.

Based on the licensee's information contained in the LAR and the above discussion in Sections 3.1 and 3.2, the NRC staff has determined that (1) LSCS EFCVs have exhibited good historical performance, and (2) that the request is consistent with existing guidance in TSTF-334, Revision 2 (Reference 3), GE- Report 004N6095 (Reference 2), and TR NEDO-32977-A (Reference 1). Therefore, the NRC staff finds the licensee's request to revise TS SR 3.6.1.3.8 to allow for a representative sample of EFCVs to be tested each refueling outage such that the test interval for each individual EFCV is approximately 10 years is acceptable and will provide reasonable assurance that high EFCV reliability will be maintained.

3.3 Evaluation of instrument lines Supporting EFCVs

During normal plant operations, EFCV position status indicators provide a timely notification to plant control room operators that a leaking or broken protection system instrument sensing line has occurred and may be contributing to the degradation of protection system functions.

Consistent with the guidance in RG 1.11, the NRC staff evaluated whether the reliability and capability of the plant operators to timely detect a degradation in protection system-related functions due to a leaking or broken instrument sensing line, will be reasonably maintained following relaxation of the EFCV position status indication surveillance frequency as described in the LAR.

As noted in Section 2.3.3 above, the LSCS UFSAR states that the provisions of RG 1.11 are incorporated into the design of the instrument sensing lines at LSCS.

According to RG 1.11, the indicating lights, associated with the position status of excess flow check valves perform an important function for instrument lines that accomplish protection functions. When an excess flow check valve has closed in response to a leak or break in an instrument sensing line, the indicating light for the valve position status will change state, thus providing timely information to control room operator that one or more protective functions may be degraded. The indicating lights for EFCVs on instrument lines serving protection functions help to minimize the development of conditions which would cause meters, annunciators, recorders, and alarms to give anomalous indications that could be confusing to the operator. Timely EFCV position status information enables operators to take appropriate preplanned actions (e.g., bypass the affected channel and promptly investigate the leak to restore protection system functionality) to mitigate this situation, and restore high functional reliability to the protection system.

LSCS UFSAR Chapter 15.6.2.1 discusses the Identification of Causes and Frequency Classification of Instrument Line Break. It describes that the EFCV position status indicating lights are used to: (a) confirm RCPB and containment integrity, and (b) alert plant control room operators that an EFCV on an instrument line serving protection system functions has closed in response to a leaking or broken instrument sensing line, which can lead to degradation in protection system functions. The NRC staff focused its evaluation on determining whether the licensee has adequately demonstrated the continued reliability of the protection system capabilities and its continued conformance with RG 1.11, after the surveillance frequency for the EFCV position indication function is extended. The staff's evaluation considered the following factors: (a) the current use of the EFCV position status indicator lights and the reliability and potential failure modes of limit/position switches in electrical circuits that are tested over significantly long surveillance intervals, and (b) possible compensating measures that could be relied on to provide additional defense-in-depth to enable the timely detection of instrument line leaks or breaks that degrade protection system functions, in the event that EFCV position status indicators are tested over significantly long surveillance intervals and subsequently fail during plant operations between surveillance intervals.

3.3.1 Reliability of EFCV Position Status Indication Capability with Extended EFCV Surveillance Frequency

As shown in several drawings (e.g., M-93 for Unit 1, M-139 for Unit 2, and others) referenced in Chapter 7 of the LSCS UFSAR (Reference 7), many of the instrument sensing lines (approximately 30 of the approximately 100 lines with EFCVs per unit) are connected to the reactor pressure vessel, the RCPB system (e.g., reactor recirculation lines), and the containment atmosphere. These instrument sensing lines are used to convey impulses representing reactor or containment performance conditions to instruments whose required function is to trip the reactor or actuate engineered safeguards.

A leak or break in an instrument sensing line will adversely impact the functionality of the protection system instruments connected to that instrument sensing line to correctly perform their required reactor protection functions or engineered safety feature (ESF) actuation functions. Specifically, in the event that a protection system instrument sensing line breaks or significantly leaks, RCPB fluids (steam) or containment air will be emitted from the broken or leaking instrument line into the secondary containment (reactor building) at a reduced (less than one gallon per minute (gpm)) flow rate, assuming the EFCV has properly performed its check function to limit the flow to accomplish its RCPB or containment atmosphere isolation function. However, if the EFCV position status indication is not functioning, depending on the size of the leak or break the resulting protection system response may or may not be observable to plant control room operators.

In some cases, a leak or break in an instrument sensing line for instruments that perform protection functions could adversely affect the capability of accomplishing one of the redundant channels of reactor protection or engineered safeguards functions. In other cases, a leak or break in an instrument line serving instruments that accomplish protection functions could cause spurious alarms and/or half-scrams or spurious ESF system half-actuations. Under certain circumstances, the effect of the leak or break could preclude the instruments in one of the redundant protection channels from reaching their calibration setpoints, thus rendering them incapable of accomplishing their required safety actions. During this time, the affected degraded channel of the reactor protection system or ESF actuation system could respond with anomalous readings that may go undetected and provide unreliable indications to plant operators.

As indicated on the LSCS UFSAR-referenced drawings, the EFCVs for the protection system-related instrument lines on each unit have been designed with position status indication capability. This capability is accomplished by a nonsafety-related system that performs safety-significant functions (e.g., reduced instruction set computer (RISC-2), because the indication plays an important role in enabling the protection system design to provide accurate, complete, and timely information pertinent to the operability status of the plant protection functions, and the operator response to the EFCV position status information minimizes the duration of conditions that can result in anomalous protection system responses. As indicated in the LAR, the proper functioning of such a position status indication system is verified through physical exercise of the EFCV during a technical specification required scheduled surveillance test, which can only be accomplished during plant outages when the protection function is not needed, and which requires periodic surveillance by simulating flow through the EFCV. After the adoption of TSTF-334, the surveillance interval will be extended from once every 2 years to once every 10 years.

Past operational experience in the nuclear industry has shown that when a switch contact, such as a manual control switch or position/limit switch, sits idle for several years, there are known failure mechanisms that can lead to the switch not properly functioning on demand. The failure mechanisms include sticking-in-place, and mis-positioning (shifting with respect to the valve piston mechanism). Therefore, such types of nuclear plant electrical circuits are usually exercised often through periodic testing (e.g., monthly, quarterly, or every refueling outage), depending on safety significance and consideration of radiological exposure to plant workers.

To confirm reliability of the EFCV position status indication capability, the NRC staff independently reviewed evidence in the LAR to determine whether the reliability of the EFCV position status indication capability will continue to be maintained after the EFCV surveillance frequency is extended. Specifically, the staff examined whether the capability of the operators

to timely detect degradation in protection functions due to instrument line leakage will continue to be maintained. To do this, the staff independently evaluated the LAR and existing design and licensing information to determine whether there is evidence to support a reasonable assurance that the capability for timely detection of reactor protection/ESF actuation degradation will be maintained.

As described above, consistent with RG 1.11 guidance, the EFCVs of approximately 30 protection system-related instrument lines on each unit at LSCS incorporate with position status indication capability to promptly alert plant operators that the instrument line may be leaking and their protection and/or safety function capabilities may be degraded. Based on the failure mechanisms for switches in electrical circuits that are not tested sufficiently often, the reduction in surveillance frequency of the position indication function of the EFCVs from once every 2 years to once every 10 years could contribute to an adverse impact on the capability of operators to readily detect a possible degradation in protection system capability. Protection systems according to GDC-21, "shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed." If the EFCV position status indication capability for an instrument line serving protection functions were to fail during the 10-year interval between successive surveillances, the ability of operators to timely detect a degraded protection system function due to a leaking or broken instrument line could be impaired unless compensating measures exist.

The EFCV surveillance test failure rate data presented by the licensee in the LAR demonstrates that the current 2-year surveillance frequency for verifying valve operation has been shown to be sufficient to achieve high functional reliability of the ability to timely detect degradation of protection functions. However, the licensee plans to increase the surveillance test interval from once per two years to once per 10 years. Therefore, to assess whether the licensee will continue to achieve high functional reliability of the ability to timely detect degradation of protection functions, the NRC staff independently evaluated whether the licensee has alternative approaches for maintaining the reliability and capability of the operators to timely detect degradation in protection functions due to instrument line leakage. The result of this evaluation is described below.

3.3.2 Evaluation of Appropriate Defense-in-Depth Compensating Measures

The NRC staff independently evaluated whether appropriate defense-in-depth compensating measures can be used to maintain sufficient reliability and capability of plant operators to timely detect broken or leaking instrument sensing lines to restore high functional reliability of the protection system. Following an evaluation of the LSCS UFSAR drawings, technical specifications, and ongoing surveillance programs covering normal operating (non-accident) plant conditions, the staff has found that the following three compensating measures serve to provide defense-in-depth for maintaining the reliability of the capability for plant operators to timely detect the degradation of protection system functions brought about by a leaking or broken instrument sensing line:

1. Frequent and regular protection system-related instrument channel checks.

   The NRC staff evaluated the LSCS TSs and determined that all or nearly all protection system-related reactor water level sensor control room indications have TS-related surveillances requiring the performance of a channel check in accordance with the licensee's SFCP. The flow-biased reactor power trip channel, which receives input from flow sensors connected to the reactor recirculation lines, is also covered by surveillance

requirements to perform channel checks. Performance of channel checks is a timely way of detecting degrading conditions including the occurrence of leaking or broken instrument sensing lines.

The NRC staff notes that many reactor pressure and reactor steam dome pressure instruments serving protection functions are not specifically covered by technical specification-related channel check surveillances. However, the staff confirmed that most of these instruments are attached to the same instrument sensing lines for which a protection function-related reactor water level sensor has a TS requirement to perform channel checks in accordance with the SFCP.

The key protection-related drywell high pressure detecting instruments do not have control room indication functions that would normally enable regular channel checks to be performed. The ability to detect broken or leaking instrument lines for these protection functions would be compensated through operator walkdowns and the enhanced surveillance methodology, as described below.

Overall, the NRC staff finds that the use of frequent channel checks of reactor water level instrument channels and flow-biased reactor power trip channels as identified in the current LSCS TSs, serve to provide defense-in-depth measures for identifying degradation in protection system functions that compensate for a reduction in surveillance frequency of the EFCV position status indication. These reactor water level instrument channel checks also serve to cover the instrument lines serving reactor pressure instrument channel functions.

2. Frequent and regular operator walkdowns of plant areas where instrument lines associated with protection functions are terminated at instrument racks and/or local pipe stands.

   The NRC staff considered the existing program for conducting regular operator rounds to ensure that safety-related equipment is in a condition that ensures that equipment will be available to perform safety functions when required. This program includes a frequent periodic walkdown of plant areas where safety-related instrument racks and pipe stands are used to support instruments performing protection system functions. If there is a broken or leaking instrument sensing line serving reactor water level or reactor pressure sensing lines there is reasonable assurance that the resulting plume of steam would be observable or the sounds of a steam leak would be audible to operators during such walkdowns. Historically, such conditions have been discovered during operator rounds at operating nuclear units, which served to alert control room operators of the onset of broken or leaking instrument sensing lines. Local area radiation monitors or ambient temperature monitors also serve to provide timely indication that a leaking or broken sensing line has occurred.

   Overall, the NRC staff finds that the current program for conducting periodic operator walkdowns of instrument racks and areas where local protection system equipment and instruments are mounted in the reactor building provides additional defense-in-depth measures for identifying degradation in protection system functions. This program helps operators identify leaking or broken instrument sensing lines, and enables timely mitigating actions to restore protection function capabilities. This program therefore serves to compensate for a reduction in surveillance frequency of the EFCV position status indication.

3. Enhanced surveillance methodologies to include a failure feedback mechanism and corrective action program to ensure that EFCV performance and reliability continues to be bounded by the topical report results.

   The NRC staff examined the LAR and the July 3, 2018, letter of approval of the licensee's RR RV-02. As described above, the NRC staff notes that TR NEDO-32977-A (Reference 1) identifies that licensees must have a failure feedback mechanism and corrective action program to ensure that EFCV performance and reliability continues to be bounded by the TR results. In the LAR, the licensee has stated that it intends to comply with the provisions of TSTF-334, and committed to evaluate any EFCV failures "to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained." In the licensee's approved RR, the licensee committed to "continue to verify the remote position indication at the same frequency as the exercise test prescribed in TS SR 3.6.1.3.8. Corrective action documents are initiated for any EFCV that fails to actuate during the program testing and for any with abnormal position indication displays." The LAR also states that "[t]o ensure EFCV performance remains consistent with the extended test interval, a minimum performance criterion has been established. The criterion specifies less than or equal to two functional failures during a 24-month rolling average to ensure that EFCV performance remains consistent with the extended surveillance interval assumptions and adverse trends in EFCV performance are identified."

   Overall, the NRC staff finds that the implementation of the enhanced surveillance processes outlined in TSTF-334 and in the licensee's LAR would serve to provide the surveillance test results and trending information that is needed to enable operators to adjust the sampling process in a manner that preserves the expected EFCV performance failure rate, and contributes to maintaining the reliability of the associated EFCV position status indication.

In summary, the NRC staff has evaluated the licensee's plant design, proposed actions described in the LAR and relief request, and the TSs and plant drawings. The staff has determined that the licensee's proposed actions in implementing TSTF-334, in conjunction with the collective defense-in-depth compensating measures, will provide reasonable assurance that the reduction in surveillance frequency will not significantly degrade the ability of plant operators to readily detect a broken or leaking instrument sensing line that is needed to support the accomplishment of protection system functions. These actions help to ensure the operator will continue to have accurate, complete, and timely information pertinent to the status of the protection functions, and will serve to enable the operator to minimize conditions which would cause meters, annunciators, recorders, and alarms to give anomalous or spurious indications.

3.4 Radiological Consequences

3.4.1 Release Frequency

The NRC staff reviewed the LSCS EFCV release frequency analysis and the consequences of the release, which are provided in GE Report 004N6095, Table 3-3 (Reference 2). The release frequency analysis for LSCS in GE Report 004N6095 follows the methodology in Sections 3.1 (failure to close) and 4.3 (release frequency) of TR NEDO-32977-A (Reference 1). Under this methodology, a postulated radiological release would occur due to an instrument line break coupled with an EFCV that fails to close after receiving the isolation signal from the main control

room. The release frequency calculations are based on a single instrument line break frequency per year and EFCV failure to close probability (EFCV unavailability). The release frequency estimates for LSCS were calculated based on a single instrument line break frequency of 5.34E-6 and an EFCV failure rate of 8.15E-7. Based on the licensees release frequency estimates, assuming 86 instrument lines with testable EFCVs, the release frequency from a broken instrument line was calculated to be approximately 1.64E-5 events/year for a 10-year surveillance test interval. For the proposed extended EFCV testing interval the dose to the public was calculated to be 8.2E-4 mrem/year (based on 0.05 rem/event). This is significantly below the 100 mrem/year NRC dose limit for individual members of the public established in 10 CFR 20.1301(a) and below the 25 mrem/year EPA dose limit in 10 CFR 20.1301(e). Because the potential dose to the public resulting from an extended EFCV testing interval continues to comply with the regulatory dose limit, the NRC staff finds the proposed extended test frequency to be acceptable with respect to release frequency.

3.4.2 Radiological Dose Assessment

The LSCS UFSAR (Reference 7), Section 15.6.2, "Instrument Line Break," assumes that the reactor is operating at design power conditions when a failure occurs in one of the instrument lines which is connected to the primary coolant system and penetrates the primary containment. The instrument line instantaneously and circumferentially breaks outside the drywell but within the secondary containment at a location where immediate detection is not automatic or apparent, and where isolation of the break might not be possible. Each line is equipped with a 0.25-inch diameter flow-restricting orifice inside the primary containment and an EFCV located outside primary containment as close as practical to the containment. Should an instrument line that forms part of the reactor coolant pressure boundary develop a leak of sufficient flow outside containment, the EFCV will close automatically. Should an EFCV fail to close when required, the 0.25-inch diameter orifice located in each line inside containment will limit flow. The current licensing basis radiological consequence analysis is based on the 0.25 inch diameter flow-restricting orifice inside the primary containment and does not credit the EFCV closure.

Further, as described in LSCS UFSAR 15.6.2.5, the instrument line break releases primary coolant to the secondary containment until the reactor is shut down and depressurized, which takes 5 hours, and is initiated 10 minutes after the occurrence of the instrument line break. No fuel damage or core uncover occurs as a result of the instrument line break. The airborne radioactivity within the secondary containment is a function of the primary coolant activity, blowdown rate, condensation rate, fraction of liquid which flashes to steam, and leakage rate from the containment. It is assumed that normal ventilation occurs for the first 10 minutes, followed by building isolation and initiation of the standby gas treatment system for the remainder of the event. The fission product activity released to the environment is based on a ventilation flow rate from the secondary containment of 110,000 cubic feet per minute (cfm) for the first 10 minutes and a standby gas treatment system iodine removal efficiency of 95 percent with a ventilation flow rate of 4,000 cfm for the duration of the accident.

The NRC has reviewed the proposed change to SR 3.6.1.3.8 against the above current licensing basis radiological consequences for the instrument line break. The NRC staff has determined that the change in the frequency of SR 3.6.1.3.8 does not affect the radiological consequences for the instrument line break because the EFCV is not credited in the radiological consequence analysis and no other changes are being proposed in this LAR. Therefore, the NRC staff finds that the proposed change does not affect the current radiological consequence analysis and concludes that the proposed changes are acceptable with respect to the radiological consequences of the instrument line break.

The NRC staff has evaluated the impact of the proposed change on the design basis instrument line break radiological consequence analysis against the regulatory requirements and guidance. The NRC staff finds, with reasonable assurance that the licensee's change to the TSs will continue to comply with the requirements of 10 CFR 20.1301(a), 10 CFR 20.1301(e), 10 CFR 100.11, and the current radiological consequence analysis. Therefore, the proposed change is acceptable with regard to the radiological consequences of the postulated instrument line break.

3.5 Technical Conclusion

Based on the above discussions in Sections 3.1 through 3.4, the NRC staff finds that the proposed changes in surveillance frequency of the EFCVs assures that the system or component function will be maintained, that facility operation will be within the Safety Limits, and that the LCO will continue to be met. The staff further finds that the extended surveillance frequency provided reasonable assurance of adequate protection of public health, safety, and security. The NRC staff's evaluation, as described in Sections 3.1 through 3.4 of this report, applies current and applicable regulatory requirements identified in Section 2.3 of this report. On this basis, the NRC staff determined that the proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment on January 22, 2019. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR, Part 20, and changes SRs. 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 (April 10, 2018 (83 FR 15415)). 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. BWROG Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000 (ADAMS Accession No. ML003729011)

2. GE Hitachi Nuclear Energy Report 004N6095, Revision 0, "LaSalle County Station Excess Flow Check Valve (EFCV) Failure Rate Analysis," dated January 2018 (ADAMS Accession No. ML18039A123).

3. Technical Specification Task Force Change TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000 (ADAMS Accession Nos. ML003775261 and ML003751245).

4. NUREG-0800, Standard Review Plan (SRP), Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, dated July 1981 (ADAMS Accession No. ML052350147).

5. Regulatory Guide (RG) 1.11, Revision 1 "Instrument Lines Penetrating the Primary Reactor Containment," dated March 2010 (ADAMS Accession No. ML100250396).

6. NUREG-1433, Revision 4, Vol. 1, "STS - GE Electrical Plant (BWR/4)," dated April 2012 (ADAMS Accession No. ML12104A192).

7. LSCS UFSAR (ADAMS Package Accession No. ML18117A015).

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Date of Issuance: February 14, 2019

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS TO RENEWED FACILITY OPERATING LICENSES RE: REVISION OF TECHNICAL SPECIFICATIONS TO ADOPT TSTF-334, "RELAXED SURVEILLANCE FREQUENCY FOR EXCESS FLOW CHECK VALVE TESTING RELATED TO EXCESS FLOW CHECK VALVE TESTING" (EPID L-2018-LLA-0035) DATED FEBRUARY 14, 2019

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ADAMS Accession No.: ML19025A288 (*) by e-mail or memo

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