Exelon Generation
4300 Winfield Road
Warrenville, IL 60555
630 657 2000 Office

10 CFR 50.55a
10 CFR 50.90

RS-18-017

February 7, 2018

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

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

Subject: Application to Revise Technical Specifications to Adopt TSTF-334-A, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," and Inservice Testing (IST) Relief Request RV-02 Related to Excess Flow Check Valve Testing Frequency

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) requests amendments to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2. The proposed changes would revise Surveillance Requirement (SR) 3.6.1.3.8 in LSCS Technical Specifications (TS) 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," to verify that a representative sample (i.e., approximately 20 percent) of reactor instrumentation line excess flow check valves (EFCVs) 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 Units 1 and 2 SFCPs currently require testing each EFCV on a 24-month frequency for SR 3.6.1.3.8.

Additionally, in accordance with 10 CFR 50.55a, "Codes and standards," paragraph (z)(1), EGC requests NRC approval of the attached relief request (RR) RV-02 associated with the fourth 10-year interval Inservice Testing (IST) Program for LSCS, Units 1 and 2. The IST RR is being submitted to request relief from the 10 CFR 50.55a requirements for testing the EFCVs in accordance with the American Society of Mechanical Engineers Operation and Maintenance (ASME OM) Code requirements. The IST RR is being submitted to modify the IST Program to be consistent with the proposed TS change.

The sampling methodology was approved by the NRC in a Safety Evaluation (SE) dated March 14, 2000, associated with the approval of the Boiling Water Reactors (BWR) Owners Group Licensing Topical Report NEDO 32977-A, "Excess Flow Check Valve Testing Relaxation." This SE also addressed the approval of Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-334-A, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," Revision 2, dated

October 31, 2000. The license amendment request (LAR) and the associated RR are consistent with the referenced SE, the associated BWR Owners Group Topical Report, and TSTF-334-A, Revision 2.

The reduced testing associated with this proposed change will result in an increase in the availability of the associated instrumentation during outages and will result in dose savings without significantly impacting the health and safety of the public while continuing to provide an acceptable level of quality and safety.

The proposed changes have been reviewed by the LSCS Plant Operations Review Committee in accordance with the requirements of the EGC Quality Assurance Program.

EGC requests approval of the proposed amendments and RR RV-02 by February 7, 2019, in support of LaSalle refueling outage L2R17 scheduled in February 2019. Once approved, the amendments shall be implemented within 45 days.

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

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 7th day of February 2018.

Respectfully,

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

Attachments:
1) Evaluation of Proposed Change
2) GE Hitachi Nuclear Energy 004N6095 R0, LaSalle County Station Excess Flow Check Valve (EFCV) Failure Rate Analysis
3) Markup of Technical Specifications Page
4) Markup of Affected TS Bases Page (For Information Only)
5) 10 CFR 50.55a Relief Request RV-02 Related to EFCV Testing Frequency

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

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

**SUBJECT:** Application to Revise Technical Specifications to Adopt TSTF-334-A, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing"

## 1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC (EGC) requests amendments to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2.

LSCS Technical Specifications (TS) 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," currently requires performance of Surveillance Requirement (SR) 3.6.1.3.8 on each excess flow check valve (EFCV) during each refueling outage. The proposed changes would revise the number of EFCVs tested by TS SR 3.6.1.3.8 from "each" to a "representative sample." The representative sample is based on approximately 20 percent of the reactor instrumentation line EFCVs such that each EFCV will be tested at least once every 10 years (nominal). Therefore, approximately 20 percent of the EFCVs will be tested every operating cycle.

The reduced testing associated with the proposed change will result in an increase in the availability of the associated instrumentation during outages and will result in dose savings. The proposed change will not significantly impact the health and safety of the public and will continue to provide an acceptable level of quality and safety.

## 2.0 DETAILED DESCRIPTION

TS SR 3.6.1.3.8 currently requires verification that each reactor instrumentation line EFCV actuates to the isolation position on an actuated or simulated instrument line break signal at a frequency in accordance with the Surveillance Frequency Control Program (SFCP). The LSCS Units 1 and 2 SFCPs currently require each EFCV to be tested on a 24-month frequency. This SR is currently implemented for the EFCVs either during the station System Leakage Testing (also known as hydrostatic testing) each outage or, for certain valves that do not require station System Leakage Testing conditions, during other outage activities.

The proposed change revises SR 3.6.1.3.8 to verify that a representative sample (i.e., approximately 20 percent) of reactor instrumentation line EFCVs actuate to the isolation position during an actual or simulated instrument line break signal, in accordance with the SFCP, such that all EFCVs will be tested at least once every 10 years (nominal). The proposed change is similar to existing performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J.

The proposed change implements Technical Specification Task Force (TSTF) Traveler TSTF-334-A, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000 (Reference 2). The proposed change is consistent with the Standard Technical Specifications (STS) changes described in TSTF-334-A, Revision 2; however, EGC proposes the following deviation:

The frequency for SR 3.6.1.3.8 as described by TSTF-334-A, Revision 2, states, "[18] months." As approved by the NRC in the LaSalle-specific Amendments 200 and 187 associated with TSTF-425 dated February 24, 2011 (Reference 4), the frequency for SR 3.6.1.3.8 states, "In accordance with the Surveillance Frequency Control Program." The frequencies allowed in the STS, NUREG-1433, Revision 4, for the comparable SR (i.e., SR 3.6.1.3.10) include:

[18] months <u>OR</u> In accordance with the Surveillance Frequency Control Program.

Because TSTF-334-A, Revision 2, was incorporated into STS, NUREG-1433, Revision 2, the use of the currently approved frequency for LSCS SR 3.6.1.3.8 (i.e., "In accordance with the Surveillance Frequency Control Program") is appropriate.

TS SR 3.6.1.3.8 will read as follows:

| | | |
|---|---|---|
| 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 |

Markups of the proposed changes to the current TS SR 3.6.1.3.8 are provided in Attachment 3.

In addition, the associated TS SR 3.6.1.3.8 Bases will be revised under the LSCS TS Bases Control Program to include implementation of TSTF-334-A, Revision 2, to provide a discussion of the basis for the test frequency and the term "representative sample." Attachment 4 provides the current affected TS Bases page with the proposed changes indicated with markups. The TS Bases pages are provided for information only and do not require NRC approval.

A previous LSCS license amendment request to implement TSTF-334-A with a corresponding relief request was submitted on May 31, 2002 (ADAMS Accession No. ML021630391). As documented in letter dated October 30, 2002 (ADAMS Accession No. ML023110393), the LSCS TSTF-334-A submittal and corresponding relief request were withdrawn.

## 3.0 TECHNICAL EVALUATION

This SR demonstrates that each EFCV is operable by verifying that the valve actuates to check flow on a simulated instrument line break downstream of the valve. The current SFCP 24-month frequency is based on the typical performance of this surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the surveillance was performed with the reactor at power. Performance of the SR for most of these valves requires the reactor to be pressurized to near normal operating pressure. The SR is implemented by isolating the affected instrument and opening a drain valve downstream of the EFCV while the process side is exposed to reactor pressure. The remaining valves have test taps, both upstream and downstream of the valve, such that, following isolation of the associated instrument, the valve may be actuated by a pressure source.

The LSCS Updated Final Safety Analysis Report (UFSAR) Section 15.6.2, "Instrument Line Break," identifies that instrumentation lines penetrating containment at LSCS from the reactor coolant pressure boundary (RCPB) are designed to high quality engineering codes and standards and seismic and environmental requirements and are equipped with a 0.25-inch diameter flow-restricting orifice inside the primary containment. Additionally, an EFCV is located outside primary containment and as close as practical to the containment. Should an instrument line that forms part of the RCPB develop a leak of sufficient flow outside containment, the EFCV will close automatically. Should an EFCV fail to close when required, a 0.25-inch diameter orifice located in each line inside containment will limit flow. UFSAR Section

15.6.2 credits the 0.25-inch orifice with limiting flow for the instrument line break accident analysis. The UFSAR Section 15.6.2 documents this event to be expected to occur with a limiting fault frequency.

The Boiling Water Reactors (BWR) Owners Group issued a topical report that provides a basis for this request. This report (NEDO 32977-A, Reference 1) provides justification for the relaxation in the SR requirement to test a representative sample of reactor instrumentation line EFCVs at the current 24-month interval required by the SFCP. The report demonstrates the high degree of EFCV reliability and the low consequences of an EFCV failure.

Several amendments have been submitted and approved for other BWRs. The format and content of these proposed TS and Bases changes are consistent with the BWR Owners Group Topical Report NEDO 32977-A and the approved generic change TSTF-334-A, Revision 2 (Reference 2), which was incorporated into NUREG-1433, Revision 2.

This evaluation reviews the following areas referenced in the NRC safety evaluation (SE), dated March 14, 2000, associated with the approval of the BWR Owners Group Topical Report NEDO 32977-A: (1) EFCV failure rate and release frequency; (2) failure feedback mechanism and corrective action program; (3) radiological dose assessment; and (4) conformance of the proposed TS to generic TSTF guidance.

### 3.1 EFCV Failure Rate and Release Frequency

The BWR Owners Group Topical Report NEDO 32977-A, dated June 2000 (Reference 1), provides detailed information about EFCV surveillance testing at 12 BWR plants. LSCS is listed as a participating utility in NEDO 32977-A; however, LaSalle data was not included as one of the 12 BWR plants listed in Table 4-1, "EFCV Failure Rates," of NEDO 32977-A.

The LSCS EFCV failure rate analysis and release frequency were evaluated in Attachment 2 of this letter in GE Hitachi Nuclear Energy Report 004N6095, Revision 0, "LaSalle County Station Excess Flow Check Valve (EFCV) Failure Rate Analysis," dated January 2018 (Reference 6).

The data provided in Table 4-1 of Attachment 2, "Summary of NEDO-32977-A EFCV Failure Rate Analysis including LSCS and Hatch," is similar to the data provided in Table 4-1 of the NEDO 32977-A Report, with the exception that Table 4-1 of Attachment 2 also includes data for the LSCS and Hatch plants. The original data from the NEDO 32977-A Report was assembled prior to the year 2000. The LSCS data included in Table 4-1 of Attachment 2 includes more recent data, including failure rate history from February 2005 through February 2017.

As stated in Attachment 2, the LSCS EFCVs are similar in design and application of those of the utilities which participated in the BWROG EFCV committee. The LSCS data were found to be consistent in both the time sampled and EFCV reliability when compared to the topical report data. A failure rate analysis of the LSCS EFCVs was done over six operating cycles, and these results show that the LSCS best estimate and upper bound failure rates are below the highest failure rates presented in NEDO-32977-A. Furthermore, the LSCS failure rates do not deviate from the industry data range or trends.

Accordingly, it is reasonable to state that the conclusions of NEDO-32977-A are applicable to the LSCS EFCV system, and that LSCS is justified in seeking the surveillance frequency relaxation.

### 3.2 Failure Feedback Mechanism and Corrective Action Program

The NRC safety evaluation associated with the BWR Owners Group Topical Report NEDO 32977-A (Reference 3) requires that each plant's corrective action program evaluate equipment failures and establish appropriate corrective actions. The LSCS Maintenance Rule (10 CFR 50.65) Program will be revised to provide a means to track the performance of the EFCVs. To 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.

### 3.3 Radiological Dose Assessment

As stated in the safety evaluation report for LSCS (NUREG-0519, Reference 5) dated March 1981, the radiological consequences for an instrument line break have been previously considered and found acceptable to the NRC. For the LaSalle instrument line break, a circumferential rupture is postulated to occur between the EFCV and the primary containment. This line leads from the reactor coolant system to an instrument rack in the secondary containment. This failure releases primary coolant to the secondary containment until the reactor is depressurized, which takes approximately five hours. The coolant flow from the break is restricted by a 0.25-inch restrictor located in the line inside the primary containment. As stated in Reference 5, the doses are less than the acceptance criteria of Section 15.6.2 of the Standard Review Plan, and a small fraction of the 10 CFR Part 100 guidelines.

The UFSAR Section 15.6.2 demonstrates (consistent with the BWROG topical report) that the failure of an EFCV has very low consequences. LSCS UFSAR Section 15.6.2 evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation does not credit the EFCV, but, instead defines the condition that the 0.25-inch diameter flow-restricting orifice is located in each line inside containment. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a five-hour period. The reactor was assumed to be operating at design power conditions prior to the break. The Standby Gas Treatment System (SGTS) and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10 CFR 100 limits. Thus, the failure of an EFCV, though not expected as a result of the proposed change, does not affect the dose consequences of an instrument line break.

This change has no impact on the UFSAR. The UFSAR, as discussed above, contains bounding evaluations of an instrument line break that is not affected by the proposed change.

### 3.4 Conformance of the Proposed TS to Generic TSTF Guidance

The proposed change to TS SR 3.6.1.3.8 is to revise the requirement to test every EFCV, by allowing a representative sample (i.e., approximately 20 percent) of EFCVs to be tested, in accordance with the Surveillance Frequency Control Program, such that all EFCVs will be tested at least once every 10 years (nominal). The term "representative sample," as proposed by the BWROG topical report and TSTF-334-A, is not defined in the TS itself.

However, the proposed Bases for SR 3.6.1.3.8 state that the representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). This is identical to the current usage in the STS, NUREG-1433, Revision 4, for SR 3.6.1.3.10. In addition, NUREG-1433, Revision 4, uses the term "representative sample" in TS SR 3.1.4.2 for verification of control rod scram times.

The Bases for SR 3.6.1.3.10 provided in TSTF-334-A, Revision 2, also includes a statement that "the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments." While the generic change includes discussions regarding grouping of the EFCVs based on valve make and environmental conditions, since all the EFCVs at LSCS are of the same make with similar environmental conditions, sub-grouping of the valves is not considered necessary at LSCS.

Changes to the LSCS Bases and the associated clarifying details of "representative sample" are subject to appropriate controls, which are specified in LSCS TS 5.5.11, "Technical Specifications (TS) Bases Control Program." Based on the low failure rate of EFCVs and the low safety significance of a failure of an EFCV (previously discussed in further detail), the level of detail in the proposed SR itself is appropriate.

## 4.0 REGULATORY EVALUATION

### 4.1 Applicable Regulatory Requirements/Criteria

STS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," of NUREG-1433, Revision 4, Standard Technical Specifications – General Electric (BWR/4) Plants provides the surveillance requirement and frequency for testing each reactor instrumentation line EFCV.

10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," provides that licensees 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 are capable of fulfilling their intended functions. The LSCS Corrective Action Program, in combination with the implementation of 10 CFR 50.65 for the EFCVs, will address actions associated with failures of EFCVs during testing.

The NRC safety evaluation (SE), dated March 14, 2000, associated with the approval of the BWR Owners Group Topical Report NEDO 32977-A, "Excess Flow Check Valve Testing Relaxation," provides the NRC's position on this topical report and industry implementation. 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 topical report results; each licensee is required to perform a plant-specific radiological dose assessment, EFCV failure analysis, and release frequency analysis to confirm that they are bounded by the generic analyses of the topical report.

BWROG topical report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000, provides (1) an estimate of steam release frequency (into the reactor building) due

to a break in an instrument line concurrent with an EFCV failure to close, and (2) an assessment of the radiological consequences of such a release.

The proposed change revises SR 3.6.1.3.8 to verify that a representative sample (approximately 20 percent) of reactor instrumentation line EFCVs actuate to the isolation position during a simulated instrument line break signal, in accordance with the SFCP, such that each EFCV will be tested at least once every 10 years (nominal). The proposed change implements Technical Specification Task Force (TSTF) Traveler TSTF-334-A, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000, with one deviation: The frequency for SR 3.6.1.3.8 as described by TSTF-334-A, Revision 2, states, "[18] months." As approved by the NRC in the LaSalle-specific Amendments 200 and 187 associated with TSTF-425 dated February 24, 2011, the frequency for SR 3.6.1.3.8 states, "In accordance with the Surveillance Frequency Control Program." This is consistent with the Standard Technical Specifications (NUREG-1433, Revision 4) for the comparable SR.

### 4.2 Precedents

Other nuclear power generating stations have received NRC approval to implement similar changes for EFCVs. The change proposed herein to the LSCS TS is consistent with these approved amendments.

1. Letter from L. N. Olshan (U.S. Nuclear Regulatory Commission) to H. L. Sumner (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB2976 and MB2977)," dated April 11, 2002 (ADAMS Accession No. ML020720594) (Reference 7)

2. Letter from R. G. Schaaf (U.S. Nuclear Regulatory Commission) to R. G. Byram (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment [Nos. 193 and 168] Re: Relaxation of Excess Flow Check Valve Surveillance Requirements (TAC Nos. MB0425 and MB0427)," dated April 11, 2001 (ADAMS Accession No. ML010960024) (Reference 8)

3. Letter from P. S. Tam (U.S. Nuclear Regulatory Commission) to J. H. Mueller (Niagara Mohawk Power Corporation), "Nine Mile Point Nuclear Station Unit No. 2 – Issuance of Amendment [No. 96] Re: Excess Flow Check Valves Surveillance Testing (TAC No. MB0301)," dated July 12, 2001 (ADAMS Accession No. ML011800115) (Reference 9)

4. Letter from G. S. Shukla (U.S. Nuclear Regulatory Commission) to D. L. Wilson (Nebraska Public Power District), "Cooper Nuclear Station – Issuance of Amendment to Revise the Technical Specifications Surveillance Test Requirement SR 3.6.1.3.8, for Excess Flow Check Valves (EFCVs) (TAC No. MB1820)," dated October 26, 2001 (ADAMS Accession No. ML01240440) (Reference 10)

### 4.3 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, "Application for amendment of license, construction permit or early site permit," Exelon Generation Company, LLC (EGC) requests amendments to Renewed Facility Operating License Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2.

The EGC LSCS, Units 1 and 2, Technical Specifications (TS) Surveillance Requirement (SR) 3.6.1.3.8 currently requires verification of the actuation capability of each reactor instrumentation line excess flow check valve (EFCV) at a frequency in accordance with the Surveillance Frequency Control Program (SFCP). The LSCS Units 1 and 2 SFCPs currently require each EFCV to be tested on a 24-month frequency. The proposed change would revise SR 3.6.1.3.8 to verify that a "representative sample" (i.e., approximately 20 percent) of reactor instrumentation line EFCVs are tested, in accordance with the SFCP, such that each EFCV will be tested at least once every 10 years (nominal). The proposed change is similar to existing performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J.

The Boiling Water Reactor Owners Group (BWROG) issued a report that provides a basis for this request. This report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000, provides justification for a relaxation in the SR frequency as described above. The report demonstrates through operating experience, a high degree of reliability with the EFCVs and the low consequences of an EFCV failure. EGC has evaluated the proposed changes against the four areas referenced in the NRC safety evaluation (SE), dated March 14, 2000, associated with the approval of the BWROG Topical Report NEDO 32977-A: (1) EFCV failure rate and release frequency; (2) failure feedback mechanism and corrective action program; (3) radiological dose assessment; and (4) conformance of the proposed TS to generic TSTF guidance.

As stated in the LSCS EFCV Failure Rate Analysis prepared by GE Hitachi (GEH) (GE Hitachi Nuclear Energy Report 004N6095, Revision 0, dated January 2018), GEH concluded that the performance of the LSCS EFCVs is comparable to the performance of those EFCVs from the utilities listed in the Boiling Water Reactors (BWR) Owners Group Licensing Topical Report NEDO 32977-A, "Excess Flow Check Valve Testing Relaxation."

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as presented below:

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

   Response: No.

   The EFCVs at LSCS, Unit 1 and Unit 2, are designed so that they will not close accidently during normal operations, will close if a rupture of the instrument line is indicated downstream of the valve, can be reopened when appropriate, and have their status indicated in the control room. This proposed change relaxes the number of EFCVs tested for TS SR 3.6.1.3.8 from "each" to a "representative sample" in accordance with the SFCP. There are no physical plant modifications associated with this change. Industry and LSCS operating experience demonstrate a high reliability of these valves. Neither EFCVs nor their failures are capable of initiating previously evaluated accidents; therefore, there can be no increase in the probability of occurrence of an accident regarding this proposed change.

   The LSCS Updated Final Safety Analysis Report (UFSAR) demonstrates, consistent with BWROG topical report NEDO-32977-A, that the failure of an EFCV has very low

consequences. The LSCS UFSAR evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation credits the 0.25-inch diameter flow-restricting orifice installed in the line with limiting flow following the instrumentation line break and does not credit the EFCV with actuating to limit leakage. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a five-hour period. The reactor was assumed to be operating at design power conditions prior to the break. The Standby Gas Treatment System (SGTS) and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10 CFR 100 limits. Thus, the failure of an EFCV, though not expected as a result of the proposed change, does not affect the dose consequences of an instrument line break.

Based on the above, it is concluded that the proposed change to the EFCV surveillance requirement does not involve a significant increase in the probability or consequences of an accident previously analyzed.

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

   Response: No.

   This proposed change allows a reduced number of EFCVs to be tested in accordance with the SFCP. The proposed change would revise SR 3.6.1.3.8 to verify that a "representative sample" (i.e., approximately 20 percent) of reactor instrumentation line EFCVs are tested, in accordance with the SFCP, such that each EFCV will be tested at least once every 10 years (nominal). No other changes in the requirements are being proposed. Industry and LSCS-specific operating experience demonstrates the high degree of reliability of the EFCVs and the low consequences of an EFCV failure. The potential failure of an EFCV to isolate by the proposed reduction in test frequency is bounded by the previous evaluation of an instrument line rupture. This change will not alter the operation or process variables, structures, systems, or components as described in the safety analysis. Thus, a new or different kind of accident will not be created from implementation of the proposed change.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The proposed changes do not involve a significant reduction in the margin of safety.

   The LSCS UFSAR evaluates a circumferential rupture of an instrument line that is connected to the primary coolant system. The evaluation credits the 0.25-inch diameter flow-restricting orifice installed in the line with limiting flow following the instrumentation line break and does not credit the EFCV with actuating to limit leakage. The dose consequences of the instrument line break are determined using the calculated mass of coolant released over approximately a five-hour period. The reactor was assumed to be

operating at design power conditions prior to the break. The SGTS and secondary containment are not impaired by the event. The evaluation concludes that the consequences of the event are well within 10 CFR 100 limits. Thus, the failure of an EFCV, though not expected as a result of the proposed change, does not affect the dose consequences of an instrument line break.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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

### 4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operations in the proposed manner, (2) 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.

## 5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendments would change a requirement with respect to installation or use of facility components located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendments do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

## 6.0 REFERENCES

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

2. Technical Specification Task Force Change TSTF-334-A, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valve Testing," dated October 31, 2000

3. Letter from S. A. Richards (U.S. Nuclear Regulatory Commission) to W. G. Warren (BWR Owners Group Chairman), "Safety Evaluation of General Electric Nuclear Energy Topical Report B21-00658-01, 'Excess Flow Check Valve Testing Relaxation,' (TAC Nos. MA7884 and M84809)," dated March 14, 2000 (ADAMS Accession No. ML003691722)

4. Letter from E. Brown (U.S. Nuclear Regulatory Commission) to M. J. Pacilio (Exelon Nuclear), "LaSalle County Station, Units 1 and 2 – Issuance of Amendments Regarding Risk-Informed Justification for the Relocation of Specific Surveillance Frequency Requirements to a Licensee-Controlled Program (TAC Nos. ME3363 and ME3364)," dated February 24, 2011 (ADAMS Accession No. ML110200143)

5. Safety Evaluation Report related to the operation of LaSalle County Station, Units 1 and 2, Docket Nos. 50-373 and 50-374, Commonwealth Edison Company, dated March 1981

6. GE Hitachi Nuclear Energy Report 004N6095, Revision 0, "LaSalle County Station Excess Flow Check Valve (EFCV) Failure Rate Analysis," dated January 2018

7. Letter from L. N. Olshan (U.S. Nuclear Regulatory Commission) to H. L. Sumner (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB2976 and MB2977)," dated April 11, 2002 (ADAMS Accession No. ML020720594)

8. Letter from R. G. Schaaf (U.S. Nuclear Regulatory Commission) to R. G. Byram (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment [Nos. 193 and 168] Re: Relaxation of Excess Flow Check Valve Surveillance Requirements (TAC Nos. MB0425 and MB0427)," dated April 11, 2001 (ADAMS Accession No. ML010960024)

9. Letter from P. S. Tam (U.S. Nuclear Regulatory Commission) to J. H. Mueller (Niagara Mohawk Power Corporation), "Nine Mile Point Nuclear Station Unit No. 2 – Issuance of Amendment [No. 96] Re: Excess Flow Check Valves Surveillance Testing (TAC No. MB0301)," dated July 12, 2001 (ADAMS Accession No. ML011800115)

10. Letter from G. S. Shukla (U.S. Nuclear Regulatory Commission) to D. L. Wilson (Nebraska Public Power District), "Cooper Nuclear Station – Issuance of Amendment to Revise the Technical Specifications Surveillance Test Requirement SR 3.6.1.3.8, for Excess Flow Check Valves (EFCVs) (TAC No. MB1820)," dated October 26, 2001 (ADAMS Accession No. ML01240440)

ATTACHMENT 2

GE Hitachi Nuclear Energy 004N6095 R0
LaSalle County Station Excess Flow Check Valve (EFCV) Failure Rate Analysis

LASALLE COUNTY STATION
UNITS 1 AND 2

Docket Nos. 50-373 and 50-374

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

13 pages follow

GE Hitachi Nuclear Energy

004N6095

Revision 0

January 2018

*Non-Proprietary Information – Class I (Public)*

# LaSalle County Station Excess Flow Check Valve (EFCV) Failure Rate Analysis

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## REVISION SUMMARY

| Rev # | Section Modified | Revision Summary |
|---|---|---|
| 0 | N/A | Initial Release |
| | | |
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# 1.0 INTRODUCTION

LaSalle County Station (LSCS) is pursuing a license amendment request and a relief request to relax testing frequency for Excess Flow Check Valves (EFCVs), utilizing NEDO-32977-A [1], which has been approved by the Nuclear Regulatory Commission (NRC). Reference 1 was commissioned by the Boiling Water Reactor Owners' Group (BWROG) with the final report being issued in June of 2000. At the time the GE Licensing Topical Report, NEDO-32977-A, was transmitted, LSCS had inconclusive EFCV data and, as a result, data concerning industry EFCV failure rates in Reference 1 did not include LSCS data. However, adequate LSCS EFCV data have since been compiled and, as detailed in this report, the design data and failure rates are comparable to the other utilities documented in Reference 1.

## 1.1 Purpose

The purpose of the LSCS EFCV failure rate analysis is not to eliminate EFCV testing, but to provide justification for relaxing EFCV testing frequency. The relaxed testing frequency allows for a representative sample of EFCVs to be tested each cycle (24 months) such that each EFCV will be tested at least once every 10 years. This essentially means that approximately 20 percent of the EFCVs will be tested every cycle.

## 1.2 Basis for Proposed Changes

The justification for the proposed changes is found in NEDO-32977-A [1], which was developed by GE for the EFCV committee of the BWROG. Reference 1 contains an upper limit failure rate which was compiled with data from 12 different stations (see NEDO-32977-A Section 4.2 and Table 4-1). LSCS data were not included in the original composite upper limit failure rate due to inconclusive EFCV failure data, and thus, the industry data contained in the report do not include LSCS plant-specific data. Though LSCS data were not included in Reference 1, it was determined that the conclusions of Reference 1 are applicable to the LSCS EFCV system, as discussed in the remainder of this report.

## 1.3 Scope

In support of a license amendment request and relief request to relax testing frequency for LSCS's EFCVs, GEH has developed this report for EFCV failure rate analysis using LSCS's plant-specific data. This report documents the results as both best estimate and upper limit failure rates. The method follows NEDO-32977-A [1], which has been approved by the NRC. The report format is similar to Hatch TSTF-334 LAR Attachment 2 to Enclosure 1, "Hatch Excess Flow Check Valve Failure Rate Results" [2]. This report also incorporates the Hatch Request for Additional Information (RAI) [3] responses into the analysis, as applicable.

Although Hatch was not one of the 12 BWR plants referenced in NEDO-32977-A (similar to LSCS), the Hatch data were found to be consistent in both the time sampled and EFCV reliability when compared to the Reference 1 data. In addition, the Hatch plant-specific EFCV failure and release rates are comparable to industry data and consistent with the NRC Safety Evaluation (SE) [4] conclusions for NEDO-32977-A.

## 2.0 LSCS PLANT-SPECIFIC EFCV DATA

The LSCS plant-specific EFCV data, similar to the data presented in the tables of NEDO-32977-A [1], are provided in Table 2-1. It should be noted that the value listed for each parameter is the same for all Unit 1 and Unit 2 EFCV models, unless otherwise specified.

The values of the parameters presented in Table 2-1 were compiled based on information from Dragon Valves EFCV drawings [5,6], as well as input from LSCS Radiation Protection (RP), Technical Information Document (IMD), and Outage Planning. These data show that the LSCS EFCVs are similar in design, and use, to the valves used by the other member utilities documented in Reference 1.

**Table 2-1: LSCS Plant-Specific EFCV Data**

| Parameter | Value |
|---|---|
| Design pressure (psig) | 1650 (model 11935-1), 1250 (model 11935-3), 45 (model 11935-5) |
| Hydrostatic pressure (psig) | 5400 |
| Make and model | Dragon Valves 11935-1, 11935-3, and 11935-5 |
| Nominal line size (inlet/outlet, inches) | 0.75/0.5 |
| Minimum flow for closure (gpm) | 6.5 |
| Test method | Offline & startup (hydro), reactor pressure, test-taps |
| Testing performed | During refueling outage and startup (hydro) |
| Testing on critical path | Yes |
| Person-Rem exposure during testing | 251 |
| Person-hours for testing during a cycle | 535 |
| Types of failures | Sticking, leak by, dirty poppet/internals |
| PM program | No |
| Failed EFCVs replaced or repaired? | Repaired |

## 3.0 FAILURE RATE ANALYSIS

This failure rate study of the LSCS EFCVs was conducted by reviewing the data over six 24 month operating cycles from February 2005 through February 2017. A total of 198 EFCVs (99 EFCVs per unit) were tested each cycle with only 10 valve failures attributed to the valve actually failing to check flow; six of these failures were associated with Unit 1 EFCVs and four of these failures were associated with Unit 2 EFCVs. The details regarding each of the EFCV failures at LSCS are provided in Table 3-1, and the operating failure rates based on this failure event data are calculated in the remainder of this section.

**Table 3-1: Summary of LSCS EFCV Failure Events**

| EFCV | Failure Year | Work Order Number | Failure Type |
|---|---|---|---|
| 1B21-F327D | 2008 | 00934549-01 | Did not go RED, did not check. Cleaned. |
| 1B21-F349 | 2008 | 00934561-01 | Valve leaks by. Repaired. |
| 1B21-F350 | 2008 | 00934561-01 | Valve leaks by. Repaired. |
| 1B21-F413A | 2010 | 01117413-01 | Failed to check, black scale type media found in valve. Cleaned poppet and valve body. Replaced spring and gasket. |
| 1B33-F305D | 2014 | 01524006-01 | Valve leaks by, minor discoloration on spring/poppet. New spring and poppet installed. |
| 1E12-F359B | 2008 | 00934478-01 | Failed to check, light corrosion found on poppet. Cleaned valve, installed new poppet, spring, gasket. |
| 2B21-F350 | 2011 | 01207808-01 | Fail to check, repaired. |
| 2B33-F307A | 2011 | 01209084-01 | Valve did not check flow. Replaced spring and poppet. |
| 2B33-F307B | 2011 | 01209084-01 | Valve failed to check. Poppet was stuck in valve body. Disassembled valve, replaced poppet and spring, reassembled valve. |
| 2B33-F317B | 2005 | 00628916-10 | Did not pass water or respond properly, inside valve and poppet a little dirty. |

The calculation of the best estimate failure rate was performed by dividing the number of failures by the total valve operating time. The calculation of the upper limit failure rate was performed using the formula listed in Section 4.2 of NEDO-32977-A [1], as follows:

$$\lambda_U = \frac{1}{2T}\chi^2_{\alpha:2r+2}$$

where:

$\lambda_U$ = the upper limit failure rate per hour

$T$ = the operating time in hours

$r$ = the number of failures

$\chi^2_{\alpha:2r+2}$ = the value taken from the chi-square distribution tables which corresponds to 2r+2 degrees of freedom and $\alpha$ = 0.05 (1 - $\alpha$ = 0.95, the specified confidence level).

For Unit 1, the operating time used was 12 years, since failure rate history was obtained from February 2005 through February 2017. Unit 1 had a total of six EFCV failures to check flow. The Chi-square value for 14 degrees of freedom and a confidence level of 0.95 is 23.7. From these values, and using the formula listed above, the upper limit failure value for Unit 1 is 1.14E-6 per hour.

A similar analysis was performed for Unit 2, spanning February 2005 through February 2017 (12 years). Unit 2 had a total of four EFCV failures to check flow. The Chi-square value for 10 degrees of freedom and a confidence level of 0.95 is 18.3. From these values, and using the formula listed above, the upper limit failure value for Unit 2 is 8.80E-7 per hour.

Additionally, a combined analysis was performed for LSCS Units 1 and 2. A summary of the LSCS EFCV failure rate analysis is provided in Table 3-2.

**Table 3-2: Summary of LSCS EFCV Failure Rate Analysis**

| Unit | Total Valve Operating Time (hours) | Failures | Best Estimate Failure Rate (per hour) | Chi-Square Value ($\chi^2$) 95% Confidence | Upper Limit of Expected Failures (per hour) |
|---|---|---|---|---|---|
| 1 | 1.04E+7 | 6 | 5.77E-7 | 23.7 | 1.14E-6 |
| 2 | 1.04E+7 | 4 | 3.85E-7 | 18.3 | 8.80E-7 |
| Total | 2.08E+7 | 10 | 4.81E-7 | 33.9 | 8.15E-7 |

The risk impact on the public health and safety from EFCVs can be evaluated as the product of a release frequency (due to a break in an instrument line concurrent with an EFCV failure to close) and the consequence of the release (Section 3.1 of NEDO-32977-A). The release frequency can be calculated based on the instrument line break frequency (Section 4.3 of NEDO-32977-A) and EFCV failure to close probability.

Table 3-3 shows the release frequency from a single instrument line, assuming different test intervals for the EFCV. The release frequency was calculated using the following equations:

$$RF = I \times \bar{A}$$

$$\bar{A} = \lambda \frac{\theta}{2}$$

where:

$RF$ = release frequency per year

$I$ = instrument line break frequency per year (Section 4.3 of NEDO-32977-A)

$\bar{A}$ = EFCV unavailability (failure to close probability)

$\lambda$ = EFCV failure rate per hour (Table 3-2)

$\theta$ = EFCV surveillance test interval in hours

**Table 3-3: Release Frequency from a Single Instrument Line (based on 5.34E-6 instrument line break frequency, and 8.15E-7 EFCV failure rate)**

| EFCV Test Interval (years) | EFCV Test Interval (hours) | EFCV Unavailability | Release Frequency (per year) |
|---|---|---|---|
| 1.5 | 13140 | 5.35E-3 | 2.86E-8 |
| 2 | 17520 | 7.14E-3 | 3.81E-8 |
| 6 | 52560 | 2.14E-2 | 1.14E-7 |
| 10 | 87600 | 3.57E-2 | 1.91E-7 |

Based on the release frequency shown in Table 3-3 for one instrument line, and assuming 86 instrument lines with testable EFCVs in a plant (consistent with NEDO-32977-A), the release frequency from any broken instrument line is:

- For 18-month surveillance test interval 86 * 2.86E-8 = 2.46E-6 events/year
- For 10-year surveillance test interval 86 * 1.91E-7 = 1.64E-5 events/year

The risk to the public can be shown by combining the release frequencies calculated above with a consequence of release (Section 3.1 of NEDO-32997-A). Consistent with the radiological consequence calculations performed in NEDO-32997-A, the corresponding public risk with the current testing basis can be shown as follows:

2.46E-6 events/year x 0.05 Rem/event = 1.2E-4 mRem/year (Whole Body)

With an extended testing interval, this value changes to the following:

1.64E-5 events/year x 0.05 Rem/event = 8.2E-4 mRem/year (Whole Body)

These values are approximately five orders of magnitude below 10CFR20.1301(a) annual exposure limits to the general public of 100 mRem/year (Whole Body). Therefore, these release frequencies are sufficiently low that it can be concluded that a change in surveillance test frequency has minimal impact on the valve reliability and radiological consequences.

## 4.0 CONCLUSIONS

Though LSCS data were not included in NEDO-32977-A [1], it was determined that the conclusions of Reference 1 are applicable to the LSCS EFCV system, as discussed below and shown in Table 4-1. Table 4-1 summarizes the results from the NEDO-32977-A EFCV failure rate analysis, as well as the results of the EFCV failure rate analysis for Hatch and LSCS.

It should be noted that LSCS was not one of the 12 BWR plants referenced in NEDO-32977-A (similar to Hatch); however, as shown in Table 4-1, the LSCS data were found to be consistent in both the time sampled and EFCV reliability when compared to the topical report data.

**Table 4-1: Summary of NEDO-32977-A EFCV Failure Rate Analysis Including LSCS and Hatch**

| Plant | Make of EFCV | Operating Time (years) | Operating Time (hours) | Number of Failures | Best Estimate Failure Rate (per hour) | Upper Limit Failure Rate (per hour) |
|---|---|---|---|---|---|---|
| Browns Ferry | Marotta | 100.5 | 8.80E+5 | 3 | 3.41E-6 | 8.81E-6 |
| Brunswick | Valcor | 267 | 2.34E+6 | 0 | 0 | 1.28E-6 |
| Clinton | Dragon | 220 | 1.93E+6 | 0 | 0 | 1.55E-6 |
| Duane Arnold (DAEC) | Marotta | 1974 | 1.73E+7 | 0 | 0 | 1.73E-7 |
| Dresden | Chemquip | 922 | 8.07E+6 | 0 | 0 | 3.71E-7 |
| Fermi 2 | Dragon | 930 | 8.15E+6 | 0 | 0 | 3.68E-7 |
| Fitzpatrick | Marotta | 2019 | 1.77E+7 | 0 | 0 | 1.69E-7 |
| Hatch | Dragon and Marotta | 783 | 6.86E+6 | 4 | 5.83E-7 | 1.33E-6 |
| LSCS | Dragon | 2376 | 2.08E+7 | 10 | 4.81E-7 | 8.15E-7 |
| Monticello | Chemquip | 2314 | 2.03E+7 | 1 | 4.93E-8 | 2.34E-7 |
| Oyster Creek | Chemquip | 465 | 4.07E+6 | 0 | 0 | 7.36E-7 |
| Susquehanna | Marotta and Valcor | 144 | 1.26E+6 | 4 | 3.17E-6 | 7.26E-6 |
| Vermont Yankee | Chemquip | 1725 | 1.51E+7 | 1 | 6.62E-8 | 3.14E-7 |
| Columbia (WNP2) | Dragon | 1344 | 1.18E+7 | 2 | 1.69E-7 | 5.34E-7 |

The manufacturer of the Unit 1 and 2 LSCS EFCVs is Dragon Valves. As shown in Table 4-1, this vendor is used by many other BWR utilities and is well represented in Reference 1. Section 2.0 of this report contains a table of LSCS specific EFCV data, similar to the tables in Reference 1. These data show that the LSCS EFCVs are similar in design, and use, to the valves used by the other member utilities. Furthermore, a failure rate analysis of the LSCS EFCVs was done over six operating cycles. As depicted in Table 4-1, these results show that the LSCS best

estimate and upper bound failure rates are below the highest failure rates presented in Reference 1, and that the LSCS failure rates do not deviate from the industry data range or trends. Section 3.0 of this report details the results of the study. This information supports the contention that the generic radiological consequences evaluation performed in Attachment B to NEDO-32977-A is applicable to LSCS. It is thus reasonable to conclude, as Reference 1 states, that similar results would be expected at LSCS.

In summary, the LSCS EFCVs are similar in design and application of those of the utilities which participate in the BWROG EFCV committee. Additionally, the LSCS failure rate study indicates that the performance of the LSCS EFCVs is comparable to the performance of those EFCVs from the utilities listed in Reference 1. Accordingly, it is reasonable to state that the conclusions of Reference 1 are applicable to LSCS and that LSCS is justified in seeking the surveillance frequency relaxation.

## 5.0 REFERENCES

1. GE Nuclear Energy, "Excess Flow Check Valve Testing Relaxation," NEDO-32977-A, June 2000.
2. Letter from H. L. Summer (Southern Nuclear Operating Company, Inc.) to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Request to Revise Technical Specifications: Excess Flow Check Valve Relaxation," HL-6105, September 19, 2001.
3. Letter from H. L. Summer (Southern Nuclear Operating Company, Inc.) to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Response to Requests for Additional Information on Technical Specification Change Request: Excess Flow Check Valve Surveillance Requirements (EFCV)," HL-6208, March 11, 2002.
4. Letter from L. N. Olshan (U.S. Nuclear Regulatory Commission) to H. L. Summer (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 andd 2 Re: Issuance of Amendments (TAC Nos. MB2976 and MB2977)," April 11, 2002.
5. Dragon Valves, Inc., "Excess Flow Check Valve," 11935-N16297, December 1975.
6. Dragon Valves, Inc., "Excess Flow Check Valve," 14531, October 1981.

ATTACHMENT 3

Markup of Technical Specifications Page

LASALLE COUNTY STATION
UNITS 1 AND 2

Docket Nos. 50-373 and 50-374

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

TS Page 3.6.1.3-8

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 ~~each~~ a representative sample of reactor instrumentation line ~~EFCV actuates~~ 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 |

**ATTACHMENT 4**

**Markup of Affected TS Bases Page**
**(For Information Only)**

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

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

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

**TS Bases Page B 3.6.1.3-14**
**INSERTS**

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.8

EFCVs are                a representative sample of

This SR requires a demonstration that ~~each~~ reactor instrumentation line ~~EFCV is~~ OPERABLE by verifying that ~~the~~ valve actuates to the isolation position on an actual or simulated instrumentation line break condition. This SR provides assurance that the reactor instrumentation line EFCVs will perform as designed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

each

INSERT 1

INSERT 2

Instrumentation lines that connect to the containment atmosphere, such as those which measure drywell pressure, or monitor the containment atmosphere or suppression pool water level, are considered extensions of primary containment. A failure of one of these instrumentation lines during normal operation would not result in the closure of the associated EFCV, since normal operating containment pressure is not sufficient to operate the valve. Such EFCVs will only close with a downstream line break concurrent with a LOCA. Since these conditions are beyond the plant design basis, EFCV closure is not needed and containment atmospheric instrumentation line EFCVs need not be tested (Ref. 6).

SR 3.6.1.3.9

The TIP shear isolation valves are actuated by explosive charges. An in place functional test is not possible with this design. The explosive squib is removed and tested to provide assurance that the valves will actuate when required. The replacement charge for the explosive squib shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of the batch successfully fired. Other administrative controls, such as those that limit the shelf life and operating life, as applicable, of the explosive charges, must be followed. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

INSERT 1

The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal).

INSERT 2

The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J.. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

ATTACHMENT 5
10 CFR 50.55a Relief Request RV-02
Related to EFCV Testing Frequency

**Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)**

## 1. ASME Code Component(s) Affected

All excess flow check valves (EFCVs) currently included in the LSCS Inservice Testing (IST) Program, as listed in the attached Table RV-02:

| Table RV-02<br>Excess Flow Check Valve Component List | | | |
|---|---|---|---|
| **VALVE NUMBER UNITS 1 and 2** | **SYSTEM** | **OM CATEGORY** | **ASME CLASS** |
| 1(2)B21-F325A | Main Steam | C | 2 |
| 1(2)B21-F325B | Main Steam | C | 2 |
| 1(2)B21-F325C | Main Steam | C | 2 |
| 1(2)B21-F325D | Main Steam | C | 2 |
| 1(2)B21-F326A | Main Steam | C | 2 |
| 1(2)B21-F326B | Main Steam | C | 2 |
| 1(2)B21-F326C | Main Steam | C | 2 |
| 1(2)B21-F326D | Main Steam | C | 2 |
| 1(2)B21-F327A | Main Steam | C | 2 |
| 1(2)B21-F327B | Main Steam | C | 2 |
| 1(2)B21-F327C | Main Steam | C | 2 |
| 1(2)B21-F327D | Main Steam | C | 2 |
| 1(2)B21-F328A | Main Steam | C | 2 |
| 1(2)B21-F328B | Main Steam | C | 2 |
| 1(2)B21-F328C | Main Steam | C | 2 |
| 1(2)B21-F328D | Main Steam | C | 2 |
| 1(2)B21-F344 | Nuclear Boiler | C | 2 |
| 1(2)B21-F346 | Nuclear Boiler | C | 2 |
| 1(2)B21-F348 | Nuclear Boiler | C | 2 |
| 1(2)B21-F350 | Nuclear Boiler | C | 2 |
| 1(2)B21-F353 | Nuclear Boiler | C | 2 |
| 1(2)B21-F355 | Nuclear Boiler | C | 2 |
| 1(2)B21-F357 | Nuclear Boiler | C | 2 |
| 1(2)B21-F359 | Nuclear Boiler | C | 2 |
| 1(2)B21-F361 | Nuclear Boiler | C | 2 |
| 1(2)B21-F363 | Nuclear Boiler | C | 2 |
| 1(2)B21-F370 | Nuclear Boiler | C | 2 |
| 1(2)B21-F372 | Nuclear Boiler | C | 2 |
| 1(2)B21-F374 | Nuclear Boiler | C | 2 |
| 1(2)B21-F376 | Nuclear Boiler | C | 2 |
| 1(2)B21-F378 | Nuclear Boiler | C | 2 |

**Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)**

**Table RV-02**
**Excess Flow Check Valve Component List**

| VALVE NUMBER UNITS 1 and 2 | SYSTEM | OM CATEGORY | ASME CLASS |
|---|---|---|---|
| 1(2)B21-413A | Reactor Core Isolation Cooling | C | 2 |
| 1(2)B21-413B | Reactor Core Isolation Cooling | C | 2 |
| 1(2)B21-415A | Reactor Core Isolation Cooling | C | 2 |
| 1(2)B21-415B | Reactor Core Isolation Cooling | C | 2 |
| 1(2)B21-F437 | Nuclear Boiler | C | 2 |
| 1(2)B21-F439 | Nuclear Boiler | C | 2 |
| 1(2)B21-F441 | Nuclear Boiler | C | 2 |
| 1(2)B21-F443 | Nuclear Boiler | C | 2 |
| 1(2)B21-F445A | Nuclear Boiler | C | 2 |
| 1(2)B21-F445B | Nuclear Boiler | C | 2 |
| 1(2)B21-F447 | Nuclear Boiler | C | 2 |
| 1(2)B21-F449 | Nuclear Boiler | C | 2 |
| 1(2)B21-F451 | Nuclear Boiler | C | 2 |
| 1(2)B21-F453 | Nuclear Boiler | C | 2 |
| 1(2)B21-F455A | Nuclear Boiler | C | 2 |
| 1(2)B21-F455B | Nuclear Boiler | C | 2 |
| 1(2)B21-F457 | Nuclear Boiler | C | 2 |
| 1(2)B21-F459 | Nuclear Boiler | C | 2 |
| 1(2)B21-F461 | Nuclear Boiler | C | 2 |
| 1(2)B21-F463 | Nuclear Boiler | C | 2 |
| 1(2)B21-F465A | Nuclear Boiler | C | 2 |
| 1(2)B21-F465B | Nuclear Boiler | C | 2 |
| 1(2)B21-F467 | Nuclear Boiler | C | 2 |
| 1(2)B21-F469 | Nuclear Boiler | C | 2 |
| 1(2)B21-F471 | Nuclear Boiler | C | 2 |
| 1(2)B21-F473 | Nuclear Boiler | C | 2 |
| 1(2)B21-F475A | Nuclear Boiler | C | 2 |
| 1(2)B21-F475B | Nuclear Boiler | C | 2 |
| 1(2)B21-F570 | Nuclear Boiler | C | 2 |
| 1(2)B21-F571 | Nuclear Boiler | C | 2 |
| 1(2)B33-F301A | Reactor Recirculation | C | 2 |
| 1(2)B33-F301B | Reactor Recirculation | C | 2 |
| 1(2)B33-F305A | Reactor Recirculation | C | 2 |
| 1(2)B33-F305B | Reactor Recirculation | C | 2 |
| 1(2)B33-F305C | Reactor Recirculation | C | 2 |
| 1(2)B33-F305D | Reactor Recirculation | C | 2 |

**Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)**

**Table RV-02**
**Excess Flow Check Valve Component List**

| VALVE NUMBER UNITS 1 and 2 | SYSTEM | OM CATEGORY | ASME CLASS |
|---|---|---|---|
| 1(2)B33-F307A | Reactor Recirculation | C | 2 |
| 1(2)B33-F307B | Reactor Recirculation | C | 2 |
| 1(2)B33-F307C | Reactor Recirculation | C | 2 |
| 1(2)B33-F307D | Reactor Recirculation | C | 2 |
| 1(2)B33-F311A | Reactor Recirculation | C | 2 |
| 1(2)B33-F311B | Reactor Recirculation | C | 2 |
| 1(2)B33-F311C | Reactor Recirculation | C | 2 |
| 1(2)B33-F311D | Reactor Recirculation | C | 2 |
| 1(2)B33-F313A | Reactor Recirculation | C | 2 |
| 1(2)B33-F313B | Reactor Recirculation | C | 2 |
| 1(2)B33-F313C | Reactor Recirculation | C | 2 |
| 1(2)B33-F313D | Reactor Recirculation | C | 2 |
| 1(2)B33-F315A | Reactor Recirculation | C | 2 |
| 1(2)B33-F315B | Reactor Recirculation | C | 2 |
| 1(2)B33-F315C | Reactor Recirculation | C | 2 |
| 1(2)B33-F315D | Reactor Recirculation | C | 2 |
| 1(2)B33-F317A | Reactor Recirculation | C | 2 |
| 1(2)B33-F317B | Reactor Recirculation | C | 2 |
| 1(2)B33-F319A | Reactor Recirculation | C | 2 |
| 1(2)B33-F319B | Reactor Recirculation | C | 2 |
| 1(2)E12-F315 | Residual Heat Removal | C | 2 |
| 1(2)E12-F317 | Residual Heat Removal | C | 2 |
| 1(2)E12-F319 | Residual Heat Removal | C | 2 |
| 1(2)E12-F359A | Residual Heat Removal | C | 2 |
| 1(2)E12-F359B | Residual Heat Removal | C | 2 |
| 1(2)E12-F360A | Residual Heat Removal | C | 2 |
| 1(2)E12-F360B | Residual Heat Removal | C | 2 |
| 1(2)G33-F312A | Reactor Water Cleanup | C | 2 |
| 1(2)G33-F312B | Reactor Water Cleanup | C | 2 |
| 1(2)G33-F309 | Reactor Water Cleanup | C | 2 |
| 1(2)E21-F304 | Low Pressure Core Spray | C | 2 |
| 1(2)E22-F304 | High Pressure Core Spray | C | 2 |

## 2. Applicable Code Edition and Addenda

The fourth 10-year interval of the LaSalle County Station (LSCS), Units 1 and 2, Inservice Testing (IST) Program is based on the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code)-2004 Edition with Addenda through OMb-2006.

## 3. Applicable Code Requirement

ISTC-3522(a), "Category C Check Valves," states, in part, "During operation at power, each check valve shall be exercised or examined in a manner that verifies obturator travel by using the methods in ISTC-5221."

ISTC-3522(c), Category C Check Valves, "If exercising is not practicable during operation at power and cold shutdowns, it shall be performed during refueling outages."

ISTC-3700, Position Verification Testing, states, in part, "Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated."

## 4. Reason for Request

In accordance with 10 CFR 50.55a, "Codes and Standards," paragraph (z)(1), relief is requested from the requirements of ASME OM Code ISTC-3522 and ISTC-3700 for the subject valves. The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety.

The ASME OM Code requires check valves to be exercised quarterly during plant operation, or if valve exercising is not practicable during plant operation and cold shutdown, it shall be performed during refueling outages.

LSCS is presently testing all EFCVs in accordance with the Surveillance Frequency Control Program (SFCP) per the LSCS Technical Specifications (TS) surveillance requirement (SR) 3.6.1.3.8 frequency (Reference 1). The LSCS Units 1 and 2 SFCPs currently require testing each EFCV on a 24-month frequency for SR 3.6.1.3.8 (References 2 and 3). A license amendment request (LAR), which is being submitted to the NRC in parallel to this request, proposes to implement Technical Specification Task Force Improved Standard Technical Specifications Change Traveler TSTF-334-A, Revision 2, by relaxing the number of EFCVs tested by TS SR 3.6.1.3.8 from "each" to a "representative sample" in accordance with the SFCP. The representative sample is based on approximately 20 percent of the reactor instrumentation line EFCVs such that each valve will be tested at least once every 10 years (nominal).

In the interim, LSCS will continue testing the EFCVs on a 24-month frequency, as documented in the SFCP, and in accordance with the IST Program requirements

pursuant to 10 CFR 50.55a. LSCS proposes, following approval of the LAR discussed above, to implement a sampling program for testing the EFCVs such that all EFCVs will be tested on a representative sampling basis (i.e., approximately 20 percent every refueling outage), and that all EFCVs will be tested at least once within a ten-year frequency.

Testing on a cold shutdown frequency is impractical considering the large number of valves to be tested and the condition, for most of the EFCVs, that reactor pressure greater than 500 pounds-per-square-inch gauge (psig) is needed for testing. In this instance, considering the number of valves to be tested and the conditions required for testing, it is burdensome to test all these valves during refueling outages. Recent improvements in refueling outage schedules minimize the time that is planned for refueling and testing activities during the outages.

The appropriate time for performing EFCV testing is during refueling outages, and for most of the EFCVs, in conjunction with the vessel hydrostatic testing. As a result of shortened outages, decay heat levels during hydrostatic tests are higher than in the past. If the hydrostatic test were extended to test all EFCVs, the vessel could require depressurization several times to avoid exceeding the maximum bulk coolant temperature limit. This is an evolution that challenges the reactor operators and thermally cycles the reactor vessel. This evolution should be avoided, if possible.

Also, based on past experience, EFCV testing during hydrostatic testing can become the outage critical path and could possibly extend a LSCS outage by two days if all EFCVs were to be tested during the refueling outage timeframe.

The testing described above requires isolation of the instruments associated with each EFCV and opening of a drain valve to actuate the EFCV. Since these instruments are in use during plant operation, isolation of any of these instruments from service may cause a spurious signal, which could result in a plant trip or an unnecessary challenge to safety systems. Additionally, process fluid will be contaminated to some degree, requiring special measures to collect flow from the drain valve and also contributes to an increase in personnel radiation exposure.

Certain EFCVs have test taps located such that reactor pressure is not required to perform these tests. In these cases, a pressure source may be applied to actuate the valve. However, this test still requires isolation of the associated plant instrumentation and could result in the plant conditions described above.

Additionally, the EFCVs have position indication at local panels, and the remote position indication is verified accurate at the same frequency as the surveillance test prescribed in TS SR 3.6.1.3.8. If the proposed request is approved, the remote position indication would continue to be verified accurate at the same frequency as the sampling program.

The EFCVs are classified as ASME Code Category C and are also containment isolation valves. However, these valves are excluded from 10 CFR 50, Appendix J, Type C leak rate testing, due to the size of the instrument lines and upstream orifices. Therefore,

they have no safety-related seat leakage criterion. Additionally, check valve remote position indication is excluded from Regulatory Guide (RG) 1.97, Revision 3, dated May 1983 (Reference 4), as a required parameter for evaluating containment isolation.

## 5. Proposed Alternative and Basis for Use

LSCS proposes to test the EFCVs on a representative sample basis at the frequency specified in the LSCS TS SR 3.6.1.3.8.

Industry experience as documented in Boiling Water Reactor (BWR) Owners Group Licensing Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," (Reference 5) indicates that EFCVs have a very low failure rate. A review of the maintenance history for LSCS EFCVs has shown that they have been extremely reliable over the life of the plant, showing less than 1 percent failure rate associated with testing of these valves. Examples of causes for the failures, as described in NEDO-32977-A, include alarm problems, position indication (limit switch adjustment), and bent instrument tubing. Review of surveillance test history at LSCS shows no evidence of time-based failure mechanisms or common mode failures associated with EFCVs. The LSCS test experience is consistent with the findings in the NEDO document. The NEDO document indicates that many reported test failures at other plants were related to test methodologies and not actual EFCV failures. Thus, the EFCVs at LSCS, consistent with the industry, have exhibited a high degree of reliability, availability, and provide an acceptable level of quality and safety.

Additionally, NEDO-32977-A states: "Failure of an EFCV to close will not involve a significant increase in the probability or consequences of an accident previously evaluated. If an EFCV fails to close, a low leak rate will exist due to the 0.25-inch orifice, effective valve restriction, or instrument tubing size. UFSAR analyses for a ruptured instrument line have shown offsite doses well below 10 CFR 100 limits without EFCVs."

NEDO-32977-A further describes the function of the EFCVs as a means to provide a simple method of avoiding adverse operational considerations associated with recovery following instrument line breaks and clearly provides that the EFCVs do not perform a function to shut down the reactor, maintain the reactor shutdown, or mitigate the consequences of an accident.

LSCS is presently testing all EFCVs in accordance with the SFCP. The SFCP requires each EFCV to be tested at least once every 24 months. A LAR, which is presently in the review process by the NRC, proposes to revise TS SR 3.6.1.3.8 to relax the SR frequency by allowing a representative sample of EFCVs to be tested in accordance with the LSCS Units 1 and 2 SFCPs.

This surveillance requirement provides assurance that the instrument line EFCVs will perform to address any operational considerations associated with an instrument line break. As described in NEDO-32977-A, the 0.25-inch orifice, effective valve restriction,

or instrument tubing size ensures that the predicted radiological consequences will not be exceeded during a postulated instrument line break event as evaluated in the LSCS UFSAR. The surveillance frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the SFCP. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

Since the EFCVs do not provide a safety function, the testing of remote position indication would not be required in accordance with the ASME OM Code. However, EGC will 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 EFCV with abnormal position indication displays.

The basis of the relief request is that the proposed alternative would provide an acceptable level of quality and safety. Therefore, this proposed alternative is requested pursuant to 10 CFR 50.55a(z)(1).

## 6. Duration of Proposed Alternative

Relief is requested for the fourth IST interval for LSCS, Units 1 and 2, which began October 12, 2017 (following approval of the LAR that is being submitted to the NRC in parallel to this request), and ends October 11, 2027.

## 7. Precedents

1. Letter from M. Markley (U.S. Nuclear Regulatory Commission) to C. R. Pierce (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 – Inservice Testing Program Relief Request and Alternatives for Pumps and Valves – Fifth Ten-Year Interval (CAC Nos. MF6238, MF6239, MF6240, MF6241, MF6242, MF6243, MF6244, MF6245, MF6246, and MF6247)," dated December 30, 2015 (ADAMS Accession No. ML15310A406) (Reference 6)

2. Letter from R. J. Laufer (U.S. Nuclear Regulatory Commission) to B. L. Shriver (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 – Third 10-Year Interval Inservice Testing (IST) Program Plans [Request No. RR-03] (TAC Nos. MC3382, MC3383, MC3384, MC3385, MC3386, MC3387, MC3388, MC3389, MC4421, MC4422)," dated March 10, 2005 (ADAMS Accession No. ML050690239) (Reference 7)

3. Letter from P. Tam (U.S. Nuclear Regulatory Commission) to J. H. Mueller (Niagara Mohawk Power Corporation), "Nine Mile Point Nuclear Station, Unit No. 2 – Authorization of Alternative [Request GVRR-08] Regarding Excess Flow Check Valve Testing Frequency (TAC No. MB1491)," dated September 17, 2001 (ADAMS Accession No. ML012340462) (Reference 8)

## 8. References

1. LaSalle Technical Specification SR 3.6.1.3.8

2. SFCP-U1, LaSalle County Station Surveillance Frequency Control Program List of Surveillance Frequencies

3. SFCP-U2, LaSalle County Station Surveillance Frequency Control Program List of Surveillance Frequencies

4. Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 3, dated May 1983

5. BWROG Topical Report, NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," June 2000

6. Letter from M. Markley (U.S. Nuclear Regulatory Commission) to C. R. Pierce (Southern Nuclear Operating Company, Inc.), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 – Inservice Testing Program Relief Request and Alternatives for Pumps and Valves – Fifth Ten-Year Interval (CAC Nos. MF6238, MF6239, MF6240, MF6241, MF6242, MF6243, MF6244, MF6245, MF6246, and MF6247)," dated December 30, 2015 (ADAMS Accession No. ML15310A406)

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