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10 CFR 50.90

RS-18-023

February 27, 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: License Amendment Request to Revise the Technical Specifications (TS) Surveillance Requirement (SR) 3.4.4.1 to Revise the Lower Setpoint Tolerances for Safety/Relief Valves (S/RVs)

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 change will modify LSCS Technical Specifications (TS) 3.4.4, "Safety/Relief Valves (S/RVs)." Specifically, the proposed amendments would expand the safety function lift setpoint tolerances for the Safety/Relief Valves (S/RVs) that are listed in the TS. This change would be limited to the lower tolerances and would not affect the upper tolerances. The tolerance band for these valves would be changed from ±3% to +3%, -5% of the setpoint.

The change is proposed in order to reduce an unnecessarily restrictive surveillance requirement. The proposed change will not impact the reliability of the S/RVs or adversely impact their ability to perform their safety function. The change will reduce the number of TS S/RV surveillance test failures for early lift pressure and preclude the submittal of previously-reportable licensee event reports (LERs) to the NRC due to setpoint drift in the low (conservative) direction.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that this change does not involve a significant hazards consideration. The bases for these determinations are included in Attachment 1 of this submittal. The historical test results of the relief valve are provided in Attachment 2 as Table 1. The proposed TS markup pages are included as Attachment 3 to this submittal. Markups of the proposed TS Bases are included for information only as Attachment 4 of this submittal.

The proposed change has 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 to the LSCS Units 1 and 2 TS by January 21, 2019, to support implementation during LaSalle refueling outage L2R17 scheduled in February 2019. Once approved, the amendments shall be implemented within 45 days. This implementation period will provide adequate time for documents to be revised using the appropriate change control mechanisms.

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 matter, 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 this 27th day of February, 2018.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager - Licensing
Exelon Generation Company, LLC

Attachments:
1) Evaluation of Proposed Change
2) Table 1, Relief Valve History
3) Markup of Technical Specifications Page
4) Markup of Affected TS Bases Pages (For Information Only)

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

ATTACHMENT 1
Evaluation of Proposed Change

SUBJECT: License Amendment Request to Revise Technical Specifications (TS) Surveillance Requirement (SR) 3.4.4.1 to Revise the Lower Setpoint Tolerances for Safety/Relief Valves (S/RVs)

1.0 SUMMARY DESCRIPTION

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

Exelon Generation Company, LLC (EGC) proposes to revise LSCS Technical Specifications (TS) 3.4, Reactor Coolant System (RCS), Section 3.4.4, "Safety/Relief Valves (S/RVs)." Specifically, EGC proposes a new safety function lift setpoint lower tolerance for the S/RVs as delineated in SR 3.4.4.1. The proposed change will revise the lower setpoint tolerances from -3 percent (%) to -5%.

This change is limited to the lower tolerances and does not affect the upper tolerances; therefore, the upper tolerance will remain at +3% of the safety function lift setpoint. In addition, this change only applies to the as-found tolerance and not to the as-left tolerance, which will remain unchanged at ±1% of the safety lift setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for S/RV testing should the tolerance be exceeded. There will be no revision to the actual setpoints of the valves installed in the plant due to this change.

The proposed change relaxes an unnecessarily restrictive SR. The proposed change will not impact the reliability of the S/RVs or adversely impact their ability to perform their safety function. This change will preclude the submittal of previously-reportable licensee event reports (LERs) to the U.S. Nuclear Regulatory Commission (NRC) due to setpoint drift in the low (conservative) direction.

2.0 DETAILED DESCRIPTION

The proposed change to LSCS, Units 1 and 2, TS 3.4.4, "Safety/Relief Valves (S/RVs)," SR 3.4.4.1, revises the safety function lift lower setpoint tolerances for the following setpoints (listed in pounds per square inch gauge (psig)):

Setpoint (psig)	Lower Tolerance
1205	Revised from -3%: - 36.1 or ≥ 1169 to -5%: - 60.2 or ≥ 1145
1195	Revised from -3%: - 35.8 or ≥ 1160 to -5%: - 59.7 or ≥ 1136
1185	Revised from -3%: - 35.5 or ≥ 1150 to -5%: - 59.2 or ≥ 1126
1175	Revised from -3%: - 35.2 or ≥ 1140 to -5%: - 58.7 or ≥ 1117
1150	Revised from -3%: - 34.5 or ≥ 1116 to -5%: - 57.5 or ≥ 1093

Note 1: This proposed license amendment request (LAR) does not make any changes to the upper setpoint tolerances; however, it would separate these values from the lower setpoint tolerances in the safety function lift setpoints listed in SR 3.4.4.1 under the "Setpoint (psig)" column. In addition, there are no changes being made to the Frequency of SR 3.4.4.1 in this LAR.

Note 2: The lower setpoint tolerance values have been rounded up to the nearest whole number.

The LSCS, Units 1 and 2 TS SR 3.4.4.1 currently reads as follows:

	SURVEILLANCE	FREQUENCY
SR 3.4.4.1	--------------------------NOTE-------------------------- Less than or equal to two required S/RVs may be changed to a lower setpoint group. -- Verify the safety function lift setpoints of the required S/RVs are as follows: Number of S/RVs — Setpoint (psig) 2 — 1205 ± 36.1 3 — 1195 ± 35.8 2 — 1185 ± 35.5 4 — 1175 ± 35.2 2 — 1150 ± 34.5 Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

The proposed TS SR 3.4.4.1 will read as follows, to include the revised lower setpoint tolerances:

	SURVEILLANCE	FREQUENCY
SR 3.4.4.1	--------------------------NOTE-------------------------- Less than or equal to two required S/RVs may be changed to a lower setpoint group. -- Verify the safety function lift setpoints of the required S/RVs are as follows: Number of S/RVs — Setpoint (psig) 2 — 1205 + 36.1, - 60.2 3 — 1195 + 35.8, - 59.7 2 — 1185 + 35.5, - 59.2 4 — 1175 + 35.2, - 58.7 2 — 1150 + 34.5, - 57.5 Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

The proposed change relaxes an unnecessarily restrictive surveillance requirement. The proposed change will not impact the reliability of the S/RVs or adversely impact their ability to perform their safety function. The S/RVs are required to meet the American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code limits based on valve type and size to ensure acceptable valve performance. These limits are not being changed. This change will preclude the submittal of certain LERs to the NRC due to setpoint drift in the low (conservative) direction.

Attachment 3 provides the markups of the proposed change to TS SR 3.4.4.1. Corresponding changes are required to the TS Bases Sections B 3.4.4.1. The TS Bases markups are provided for informational purposes in Attachment 4. Upon approval of this LAR, the TS Bases will be revised in accordance with the LSCS Units 1 and 2 TS Bases Control Program.

3.0 TECHNICAL EVALUATION

3.1 Background

The nuclear pressure relief system consists of S/RVs located on the main steam (MS) lines between the reactor vessel and the first isolation valve within the drywell. These valves protect against overpressure of the reactor coolant system (RCS).

The S/RVs provide three main protection functions:

a. Overpressure relief operation - The valves open automatically to limit a pressure rise.

b. Overpressure safety operation - The valves function as safety valves and open (self-actuated operation if not already automatically opened for relief operation) to prevent nuclear system overpressurization.

c. Depressurization operation - The automatic depressurization system (ADS) valves open automatically as part of the emergency core cooling system (ECCS) for events involving small breaks in the nuclear system process barrier (reactor coolant pressure boundary).

In the safety mode, or the spring mode of operation, the valves open when steam pressure at the valve inlet overcomes the spring force holding the valve closed. This mode satisfies the ASME Code requirements. It is this mode of operation for which the lower surveillance tolerances for the safety function as-found lift setpoints will be relaxed from -3% to -5%. The as-found upper surveillance tolerances will remain at +3%. The relief and automatic depressurization modes rely upon solenoid actuation to open the valves and are not affected by this proposed change.

The LSCS S/RVs are Crosby Style 6xRx10 HB-65-BP S/RVs. The LSCS S/RVs were originally purchased to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Class 1, 1971 Edition, Summer 1972 Addenda including Code Cases 1567 and 1711.

A review of as-found test data for the LSCS S/RVs indicates a tendency for minor setpoint drift in the negative direction. LSCS experience shows that it is the nature of these valves to have a drift/variance with an initial as-found low lift pressure. During some test sequences, one or two

S/RVs out of a sample size of six initially lift between -5% to -3%. The second and third test lifts following the initial test lift will typically be within the ±3% band.

Currently, at least six S/RVs are removed during each refueling outage, bench tested for safety set pressure and replaced with valves certified to have zero seat-to-disk leakage and to have safety lift setpoint tolerances within ±1% of the setpoint as specified in the TS SR 3.4.4.1. If the as-found lift is greater than the ±3% tolerance for one of the S/RVs tested from the original sample size, the sample size will be increased by two S/RVs, in accordance with the inservice testing (IST) Program requirements.

On March 8, 1993, the NRC issued a Safety Evaluation (SE) for the GE Nuclear Energy Licensing Topical Report (LTR) NEDC-31753P submitted by the Boiling Water Reactor Owners Group (BWROG) (Reference 6.1). In the SE, the NRC stated that a generic change of setpoint tolerance to ±3% is acceptable provided that it is evaluated in the analytical bases. The required analysis was completed for LSCS and the change was approved by the NRC (References 6.2 and 6.3).

The operability of the S/RVs is based on the TS SR acceptance criteria with a setpoint tolerance of ±3%. If any S/RV exceeds the tolerance, an Issue Report (IR) for each S/RV that exceeds the tolerance is entered into the LSCS Corrective Action Program to evaluate the test failure. In addition, test failures outside of ±3% would result in testing additional valves to comply with the ASME OM Code requirements.

3.2 Evaluation

The proposed lower setpoint tolerance change from -3% to -5%, was evaluated using the previously accepted methodology of the LTR and the associated SER. Since the evaluation performed in detail to support the current upper and lower tolerances of ±3%, and the conclusions of the evaluation have not changed, only those areas not previously reviewed by the NRC are included in this evaluation.

3.2.1 Thermal Limits

The effect of adjusting the lower S/RV tolerance from -3% to -5% on Anticipated Operational Occurrences (AOOs) has been evaluated for the Operating Limit Minimum Critical Power Ratios (MCPRs), off-rated MCPRs, and off-rated Linear Heat Generation Rates (LHGRs). All AOO thermal limits analyses credit the relief mode operation of the S/RV. The S/RV safety function pressure setpoints, expanded to the proposed -5%, are still greater than the S/RV relief function pressure setpoints assumed in the analyses; thus, the time at which the S/RVs lift in the analyses is unchanged. Therefore, there is no adverse impact on the AOO thermal limits as the current AOO thermal limits bound the expanded S/RV range (Reference GE Hitachi Nuclear Energy (GEH) Report 004N6801-R0 – Reference 6.6).

3.2.2 ASME Overpressure

A postulated Main Steam Isolation Valve (MSIV) fast closure is the limiting event for peak vessel pressurization for ASME overpressure consideration, where the vessel dome (and thus bottom) pressure is maximized. In any pressurization case, an earlier opening of the S/RVs due to lower setpoints would produce improved results and gain margin. Analysis of the ASME Overpressure peak pressure event is conservatively based on the S/RV upper tolerance

settings (i.e., +3% setting). Therefore, lowering the S/RV tolerance range from -3% to -5% does not have an adverse impact on the ASME overpressure analysis result.

3.2.3 LOCA Analysis

The Loss of Coolant Accident (LOCA) credits the ADS function of the S/RVs during a small break. The ADS function is not based on pressure setpoints. During a large break LOCA the reactor vessel is automatically depressurized due to the nature of the event and the S/RV safety function is not actuated (Reference GE Nuclear Energy Report GE-NE-B13-01760 – Reference 6.7). Therefore, in any type of LOCA there is no adverse impact from lowering the S/RV tolerance range from -3% to -5%.

3.2.4 High Pressure System Performance

The high pressure systems include Reactor Core Isolation Cooling (RCIC) and High Pressure Core Spray (HPCS) as well as Standby Liquid Control (SLC) system. The most significant effect of changing the S/RV setpoint tolerance on the HPCS and RCIC and SLC systems' operations is the maximum reactor pressure at which they are required to deliver flow to the reactor. Since the limiting safety analyses are conservatively based on the S/RV upper tolerance settings (i.e., +3% setting), lowering the S/RV tolerance range from -3% to -5% does not have an adverse impact on the high pressure system performance.

3.2.5 ATWS Mitigation (Peak Pressure Suppression Pool Temperature)

The limiting Anticipated Trip Without Scram (ATWS) case for peak reactor pressure is similar to the ASME Overpressure event (MSIV closure) discussed above, except that a complete failure of scram is assumed. In any pressurization case, an earlier opening of the S/RVs due to lower setpoints would produce improved results and gain margin. Analysis of the ATWS peak pressure and suppression pool temperature are conservatively based on the S/RV upper tolerance settings (i.e., +3% setting). Therefore, lowering the S/RV tolerance range from -3% to -5% does not have an adverse impact on the ATWS overpressure or suppression pool temperature analysis results.

3.2.6 Containment Analyses

Containment analyses such as peak suppression pool temperature, peak suppression pool pressure, discharge line dynamic load, and submerged structure loads are all most limited by high mass/energy flow rates from the vessel into the containment structure. The flow rate through the S/RVs is directly proportional to the pressure differential across them. Since allowing lower pressure differences at the minimum end does not affect the limiting analyses at the maximum end of the pressure range, there is no impact on the containment systems, structures, or components' ability to manage energy release during S/RV actuation. Current containment analyses bound the proposed change (Reference GE Nuclear Energy Report GE-NE-B13-01760 – Reference 6.7).

3.3 Operating Margin

The purpose of the lower setpoint tolerance is to ensure sufficient margin exists between the normal operating pressure of the system and the point at which the S/RVs actuate in the overpressure safety mode. The nominal operating pressure of the reactor pressure vessel at

power is 1050 psig. A lower setpoint tolerance value of -5%, applied to the lowest nominal S/RV (1150 psig) would allow it to lift at 1093 psig. The lowest potential margin between the nominal reactor pressure and the S/RV lift pressure of the valves with the lowest overpressure safety setpoint is 43 psig. Also, the relief mode for two S/RVs is set at 1076 psig and four S/RVs at 1086 psig. The relief mode on these S/RVs would actuate before the above stated 1093 psig lowest allowable lift. The potential margin is sufficient to prevent unwanted actuation of the S/RVs, since the relief mode setpoint ensures that pressure transients will still cause the valves to open in the relief mode prior to the safety mode.

The Crosby Valve and Gage Company Procedure I-11069, "Instruction Manual for Crosby Style 6xRx10 HB-65-BP Safety Relief Valve for Main Steam Service," Revision 1, (Reference 6.4) discusses valve performance in Section 3. Step 3.1.1.4 states the following:

> Setpoint repeatability for on line service should be within a tolerance of +1, -3% of nameplate set pressure. However, the low limit of the setpoint tolerance may be extended to 1067 psig if all other valve functional requirements are met.

The requested SR 3.4.4.1 revision will lower the minimum S/RV as-found set point acceptance criteria for the lowest setpoint from 1116 psig to 1093 psig. The SR 3.4.4.1 minimum acceptance criteria will still be greater than the manufacturer's recommended low setpoint pressure limit of 1067 psig.

The proposed revision to SR 3.4.4.1 only changes the S/RV setpoint acceptance criteria minimum tolerance. The as-left or as-installed setpoint tolerances are unchanged; therefore, the revision does not affect the actual operation of the S/RVs. The requested SR 3.4.4.1 revision has no effect on the S/RV design basis such as simmering, seat leakage, or valve reliability. As previously discussed, the revised setpoint tolerance will still exceed the manufacturer's recommended minimum tolerance for S/RV as-found setpoint testing.

The valves removed for testing are returned to a tolerance of ±1% prior to being installed for service, thereby returning the margin to the original levels. Therefore, the margin is considered adequate and will not impact normal plant operation.

3.4 Surveillance Test History

Test results for LSCS S/RVs have shown that approximately 11% of S/RVs experience minor setpoint drift sufficient to exceed the acceptance criteria of -3% in the negative direction over time. Based on a review of the 133 previous test results shown in Table 1 for the LSCS valves (included in Attachment 2 of this submittal), the average drift was -1.32%. There were 14 valve tests that failed below the -3% tolerance with three of these valves testing below -5%. The data includes tests for valves that span up to 6 years between bench tests. Therefore, excessive drift over six years is not anticipated for the S/RVs. The S/RV leakage is monitored and determined by the S/RV tailpipe temperatures recorded in the main control room. If a valve's tailpipe temperature reaches a specified temperature, the S/RV is considered to be leaking and will typically be replaced at the next refueling outage. The test results shown in Table 1 provides the last three tests for each of the S/RVs, if available. This covers a period from refueling outage 02 in 1988 to refueling outage 16 in 2017 for Unit 2 and refueling outage 02 in 1988 to refueling outage 16 in 2016 for Unit 1.

The LSCS test data review identified nine of the 14 test failures described above occurred prior to the Unit 1 and Unit 2 Cycle 10 outages, on January 13, 2004, and February 7, 2005, respectively. A modification to the S/RVs was performed as discussed in LER 2003-003-00 submitted May 8, 2003 (Reference 6.5). In this LER, a common failure mechanism was identified in which seat leakage resulted in a negative setpoint drift. The corrective action was to replace the semi-flex disc design with a full-flex design to eliminate seat leakage. Following the modification, the as-found valve failures have been drastically reduced and the subsequent failures have been identified to be the result of a typical expected setpoint drift. Any subsequent repairs performed on these valves were required as a result of expected minor wear and not due to a common failure mechanism.

4.0 REGULATORY EVALUATION

LSCS Updated Final Safety Analysis Report (UFSAR) Sections 5.2, "Integrity of Reactor Coolant Pressure Boundary," and 3.1, "Conformance with NRC General Design Criteria," provide detailed discussion of LSCS's compliance with the applicable regulatory requirements and guidance.

The proposed TS amendments:

a. Do not alter the design or function of any system;

b. Do not result in any change in the qualifications of any component; and

c. Do not result in the reclassification of any component's status in the areas of shared, safety-related, independent, redundant, and physically or electrically separated.

4.1 Applicable Regulatory Requirements/Criteria

4.1.1 Applicable 10 CFR 50 Appendix A General Design Criteria (GDC)

The following GDCs for the RCS, which require that the system be protected from overpressurization, were evaluated to determine if these GDC continue to be met.

GDC 15 – Reactor Coolant System Design (Criterion 15)

The reactor coolant system (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).

An example of the integrated protective action scheme, which provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded, is the automatic initiation of the nuclear system pressure relief system upon receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided to discharge steam from the nuclear system to the suppression pool. The nuclear system pressure relief system also provides for automatic depressurization of the nuclear system in the event of a loss of coolant accident (LOCA) in which the vessel is not depressurized by the accident. The depressurization of the nuclear system in this situation allows operation of the low pressure ECCS to supply enough cooling water to adequately cool the core. In a similar

manner, other auxiliary, control, and protection systems ensure that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including AOOs. Criterion 15 is not affected by this proposed change.

GDC 35 – Emergency Core Cooling (Criterion 35)

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The ADS functions to reduce the reactor pressure so that flow from Low Pressure Coolant Injection (LPCI) and Low Pressure Core Spray (LPCS) enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the nuclear system pressure relief valves to relieve the high-pressure steam to the suppression pool. Criterion 35 is not affected by this proposed change.

Thus, the proposed change to the S/RV lower setpoint tolerances does not change the conformance with the above GDC and these GDC will continue to be met by this LAR.

4.1.2 Applicable ASME Code Requirements

The LSCS IST program is currently implemented in accordance with the requirements of the ASME OM Code 2004 Edition through OMb-2006 Addenda. The S/RVs at LSCS are Class 1 Category C valves in accordance with the LSCS IST Program. As required by the ASME OM Code, additional valves would be tested if the as-found setpoint of a tested valve from the sample exceeds ±3% of the nameplate set pressure. If one of these additional valves fails, then all of the main steam S/RVs would be removed and tested. OM Code testing of all S/RVs is accomplished over a period of five years. The LSCS S/RVs will continue to be tested in accordance with these requirements.

The ASME OM Code Mandatory Appendix I, Inservice Testing of Pressure Relief Devices in Light-Water Reactor Nuclear Power Plants, Guiding Principles, paragraph I-1310(e), *Acceptance Criteria*, allows the owner to establish setpoint acceptance criteria for relief valves tested under the IST Program; therefore, no relief will be required with regard to the setpoint tolerance change from -3% to -5%. However, a change to the LSCS TS will be required and is being proposed with this LAR.

4.2 Precedent

Additional nuclear power generating stations have received NRC approval to implement similar amendments for S/RVs, such as Columbia Generating Station, Susquehanna Steam Electric Station, and River Bend Station, as delineated below. The change proposed herein to the LSCS TS is consistent with these approved amendments.

1. Letter from L. J. Klos (U.S. Regulatory Commission) to M. E. Reddemann (Energy Northwest), "Columbia Generating Station – Issuance of Amendment Re: To Modify Technical Specifications Surveillance Requirements 3.4.3.1 and 3.4.4.1 Safety/Relief Valve Setpoint Lower Tolerance (CAC No. MF7699)," dated March 9, 2017 (ADAMS Accession No. ML17052A125)

2. Letter from B. K. Vaidya (U.S. Regulatory Commission) to T. S. Rausch (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendments Re: Change to Technical Specifications (TSs) Surveillance Requirements (SRs) 3.4.3.1 to Revise the Lower Surveillance Tolerances (TAC Nos. ME5050 and ME5051)," dated November 17, 2011 (ADAMS Accession No. ML11292A137)

3. Letter from M. Webb (U.S. Nuclear Regulatory Commission) to P. D. Hinnenkamp (Entergy Operations, Inc.), "River Bend Station, Unit 1 – Issuance of Amendment Re: Modification of the Technical Specification Surveillance Requirements for the Safety/Relief Valves (TAC No. MB5090)," dated February 13, 2003 (ADAMS Accession No. ML030450307)

4.3 No Significant Hazards Consideration

Amendments are proposed to the LaSalle County Station (LSCS), Units 1 and 2 Technical Specification (TS) surveillance requirement (SR) 3.4.4.1, "Safety/Relief Valves (S/RVs)," to change the lower surveillance lift setpoint tolerance for the S/RVs. The current tolerance is based on ±3%. The proposed change will revise the lower setpoint surveillance tolerances from -3% to -5%. This change only applies to the as-found tolerance and not to the as-left tolerance, which will remain unchanged at ±1%, as specified in TS SR 3.4.4.1, of the safety lift setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for S/RV testing. No changes to the actual safety function lift setpoints are required for the valves installed in the plant. There will be no physical modifications to the valves.

Exelon Generating Company, LLC (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. **Do the proposed amendments involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

   This change has no influence on the probability or consequences of any accident previously evaluated. The lower setpoint tolerance change does not affect the operation of the valves and it does not change the as-left setpoint tolerance. The change only affects the lower tolerance for valve opening and does not change the upper tolerance, which is the limit that protects from overpressurization.

   The proposed amendments do not involve physical changes to the valves, nor do they change the safety function of the valves. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions and no changes to existing structures, systems, or components.

The proposed amendments do not change any other behavior or operation of any safety/relief valves (S/RVs), and, therefore, has no significant impact on reactor operation. They also have no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the Updated Final Safety Analysis Report (UFSAR).

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

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

   Response: No.

   The proposed change to the S/RV safety lower setpoint tolerance from -3% to -5% only affects the criteria to determine when an as-found S/RV test is considered to be acceptable. This change does not affect the criteria for the upper setpoint tolerance.

   The proposed lower setpoint tolerance change does not adversely affect the operation of any safety-related components or equipment. The proposed amendments do not involve physical changes to the S/RVs, nor do they change the safety function of the S/RVs. The proposed amendments do not require any physical change or alteration of any existing plant equipment. No new or different equipment is being installed, and installed equipment is not being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated. This change does not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no changes are being made to the procedures relied upon to respond to an off-normal event as described in the UFSAR. As such, no new failure modes are being introduced. The change does not alter assumptions made in the safety analysis and licensing basis.

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

3. **Do the proposed amendments involve a significant reduction in a margin of safety?**

   Response: No.

   The proposed lower setpoint tolerance change only affects the criteria to determine when an as-found S/RV test is considered to be acceptable. This change does not affect the criteria for the S/RV setpoint upper setpoint tolerance. The TS setpoints for the S/RVs are not changed. The as-left setpoint tolerances are not changed by this proposed change and remain at ±1% of the safety lift setpoint.

   The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to

an event. The proposed change does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation.

Therefore, this proposed change does 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 Conclusions

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by 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 a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed 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

6.1. GE Nuclear Energy Licensing Topical Report NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Revision Licensing Technical Report," dated February 1990

6.2 Letter from W. R. Butler (U.S. Nuclear Regulatory Commission) to D. L. Farrar (Commonwealth Edison Company), "Issuance of Amendment No. 28 to Facility Operating License No. NPF-11 and Amendment No. 15 to Facility Operating License No. NPF 18 – LaSalle County Station, Units 1 and 2," dated October 31, 1985 (ADAMS Accession No. ML021120118)

6.3 Letter from R. M. Latta (U.S. Nuclear Regulatory Commission) to D. L. Farrar (Commonwealth Edison Company), "Issuance of Amendment [No. 108] (TAC No. M93915)," dated January 3, 1996 (ADAMS Accession No. ML021130181)

6.4 Crosby Valve & Gage Company, Procedure No. I-11069, Revision 1, Instruction Manual for Crosby 6xRx10 HB-65-BP Safety Relief Valve for Main Steam Service, Approval Date June 17, 1982

6.5 Letter from S. Landahl (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission), "Licensee Event Report," dated May 8, 2003

6.6 GE Hitachi Nuclear Energy (GEH) Report 004N6801-R0, "Evaluation of LaSalle Spring Safety Valve Opening Setpoint Uncertainty Extension," dated February 8, 2018

6.7 General Electric Nuclear Energy Report GE-NE-B13-01760, "Safety Review for LaSalle County Station Unit 1 and 2 Safety/Relief Valves Reduction and Setpoint Tolerance Relaxation Analyses," dated March 1995

Table 1
Relief Valve History

Valve Number	Install Unit	Outage/ Year Tested	Set Pressure (psig)	As-Found Set Pressure (psig)	Deviation (%)	Comments
1	1	R16/2016	1150	1150	0	
	2	R13/2011	1150	1165	1.3	
	2	R10/2005	1150	1143	-0.609	
2	2	R16/2017	1150	1152	0.174	
	1	R15/2014	1150	1135	-1.39	
	2	R12/2009	1150	1158	0.696	
3	1	R15/2014	1175	1172	-0.255	
	2	R14/2013	1175	1170	-0.426	
	1	R11/2006	1175	1174	-0.085	
4	2	R16/2017	1175	1180	0.426	
	2	R12/2009	1175	1180	0.426	
	2	P01/2002	1175	1139	-3.06	Failed > -3% and < -5% LER-2003-003-00
5	1	R16/2016	1175	1195	1.7	
	2	R13/2011	1175	1149	-2.21	
	2	R10/2005	1175	1166	-0.766	
6	1	R12/2008	1175	1147	-2.38	
	2	F36/2003	1175	N/A	N/A	Excessive Leakage After SCRAM
	1	R09/2002	N/A	N/A	N/A	Could Not Test - Damaged Studs
	2	R07/1996	1175	1187	1.02	
7	1	R16/2016	1185	1200	1.26	
	2	R13/2011	1185	1187	0.169	
	1	R10/2004	1185	1200	1.26	
8	1	R15/2014	1175	1170	-0.426	
	1	F40/2011	1175	N/A	N/A	Replaced With New Valve
9	1	R16/2016	1185	1177	-0.68	
	2	R13/2011	1185	1188	0.253	
	1	R10/2004	1185	1198	1.1	

Table 1 Relief Valve History						
Valve Number	Install Unit	Outage/ Year Tested	Set Pressure (psig)	As-Found Set Pressure (psig)	Deviation (%)	Comments
10	2	R07/1996	1185	1209	2.02	
	1	R06/1994	1185	1163	-1.86	
	2	R02/1988	1185	1056	-10.89	Failed > -5%
11	1	R12/2008	1175	1184	0.766	
	2	P01/2002	1175	1113	-5.28	Failed > -5% LER-2003-003-00
	1	R08/1999	1175	1145	-4.18	Failed > -3% and < -5%
12	2	R16/2017	1195	1130	-5.439	Failed > -5%
	1	R14/2012	1195	1205	0.836	
	2	R11/2007	1195	1191	-0.335	
13	1	R15/2014	1195	1198	0.251	
	2	R12/2009	1195	1192	0.251	
	2	P01/2002	1195	1208	1.09	
14	2	R15/2015	1195	1145	-4.18	Failed > -3% and < -5%
	1	R13/2010	1195	1215	1.67	
	2	R10/2005	1195	1196	0.0837	
15	1	R15/2014	1205	1154	-4.23	Failed > -3% and < -5%
	2	R12/2009	1205	1208	0.249	
	2	P01/2002	1205	1149	-4.65	Failed > -3% and < -5% LER-2003-003-00
16	2	R14/2013	1205	1222	1.41	
	1	R11/2006	1205	1193	-0.996	
	2	R08/2000	1205	1169	-2.99	
17	1	R13/2010	1205	1208	0.249	
	1	R10/2004	1205	1214	0.747	
	2	R08/2000	1205	1213	0.664	
18	2	R15/2015	1150	1099	-4.43	Failed > -3% and < -5%
	2	R14/2013	1150	1136	-1.22	
	1	R11/2006	1150	1147	-0.261	

Table 1
Relief Valve History

Valve Number	Install Unit	Outage/ Year Tested	Set Pressure (psig)	As-Found Set Pressure (psig)	Deviation (%)	Comments
18	2	R08/2000	1205	1171	-2.82	Removed from plant
	2	R07/1996	1205	1188	-1.41	
	1	R06/1994	1205	1186	-1.58	
63	2	R16/2017	1175	1167	-0.681	
	1	R14/2012	1175	1180	0.426	
	2	R11/2007	1175	1172	-0.255	
64	1	R12/2008	1150	1151	0.087	
	2	P01/2002	1150	1129	-1.83	
	1	R08/1999	1150	1135	-1.3	
65	1	R16/2016	1205	1234	2.41	
	2	R13/2011	1205	1214	0.747	
	2	R10/2005	1205	1216	0.913	
66	2	R16/2017	1205	1206	0.083	
	1	R14/2012	1205	1205	0	
	2	R11/2007	1205	1194	-0.913	
67	2	R16/2017	1195	1182	-1.088	
	1	R13/2010	1195	1201	0.502	
	1	R10/2004	1195	1210	1.26	
68	1	R14/2012	1185	1212	2.28	
	2	R11/2007	1185	1157	-2.36	
	1	R09/2002	1185	1148	-3.12	Failed > -3% and < -5%
69	1	R16/2016	1195	1160	-2.93	
	2	R13/2011	1195	1194	-0.084	
	2	R10/2005	1195	1205	0.837	
70	1	R09/2002	1205	1170	-2.9	
	1	F35/1996	1205	1201	-0.332	
	1	R06/1994	1205	1195	-0.83	

Table 1
Relief Valve History

Valve Number	Install Unit	Outage/ Year Tested	Set Pressure (psig)	As-Found Set Pressure (psig)	Deviation (%)	Comments
71	2	R16/2017	1185	1191	0.506	
	1	R14/2012	1185	1183	-0.169	
	2	R11/2007	1185	1171	-1.18	
72	1	R09/2002	1205	1192	-1.08	
	2	R07/1996	1205	1231	2.16	
	1	R05/1992	1205	1175	-2.49	
73	1	R16/2016	1175	1210	2.98	
	2	R13/2011	1175	1178	0.255	
	2	R10/2005	1175	1167	-0.681	
74	2	R14/2013	1175	1183	0.681	
	1	R11/2006	1175	1173	-0.17	
	2	R08/2000	1175	1143	-2.72	
75	1	R09/2002	1185	1157	-2.36	
	2	R07/1996	1185	1173	-1.01	
	2	R04/1992	1185	1196	0.93	
76	1	R13/2010	1150	1129	-1.83	
	1	R10/2004	1150	1160	0.087	
	1	R08/1999	1150	1111	-3.39	Failed > -3% and < -5%
77	2	R16/2017	1175	1131	-3.745	Failed > -3% and < -5%
	1	R14/2012	1175	1172	-0.255	
	2	R11/2007	1175	1141	-2.89	
78	2	R14/2013	1195	1191	-0.335	
	1	R11/2006	1195	1192	-0.251	
	1	R08/1999	1195	1181	-1.17	
79	1	R15/2014	1195	1198	0.251	
	2	R12/2009	1195	1205	0.837	
	2	P01/2002	1195	1209	1.17	

Table 1
Relief Valve History

Valve Number	Install Unit	Outage/ Year Tested	Set Pressure (psig)	As-Found Set Pressure (psig)	Deviation (%)	Comments
80	2	R16/2017	1195	1200	0.418	
	1	R12/2008	1195	1197	0.167	
	2	P01/2002	1195	1209	1.17	
102	2	R15/2015	1205	1208	0.249	
	1	R12/2008	1205	1213	0.664	
	2	P01/2002	1205	1159	-3.82	Failed > -3% and < -5% LER-2003-003-00
103	1	R09/2002	N/A	N/A	N/A	No Test – Damaged Studs
	2	R07/1996	1195	1190	-0.418	
	1	R05/1992	1195	1175	-1.67	
104	2	R16/2017	1150	1169	1.652	
	1	R14/2012	1150	1165	1.3	
	2	R11/2007	1150	1147	-0.261	
105	2	R08/2000	1150	1152	0.174	
	2	R04/1992	1150	1158	0.7	
	1	R02/1988	1150	1140	-0.87	
106	2	R15/2015	1175	1171	-0.34	
	2	R12/2009	1175	1196	1.79	
	2	P01/2002	1175	1127	-4.08	Failed > -3% and < -5% LER-2003-003-00
107	2	R14/2013	1195	1194	-0.084	
	1	R11/2006	1195	1175	-1.67	
	2	R08/2000	N/A	N/A	N/A	Replaced in R08
108	2	R15/2015	1175	1184	0.766	
	1	R13/2010	1175	1170	-0.426	
	1	R10/2004	1175	1182	0.596	
109	2	R15/2015	1185	1181	-0.338	
	1	R12/2008	1185	1178	-0.591	
	2	P01/2002	1185	1198	1.1	
110	2	R15/2015	1185	1188	0.253	
	1	R12/2008	1185	1193	0.675	
	2	P01/2002	1185	1182	-0.253	

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 3.4.4-2
INSERT

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	------------------NOTE------------------ Less than or equal to two required S/RVs may be changed to a lower setpoint group. -- Verify the safety function lift setpoints of the required S/RVs are as follows: Number of S/RVs — Setpoint (psig) 2 — ~~1205 ± 36.1~~ 3 — ~~1195 ± 35.8~~ 2 — ~~1185 ± 35.5~~ 4 — ~~1175 ± 35.2~~ 2 — ~~1150 ± 34.5~~ Replace with INSERT Following testing, lift settings shall be within ± 1%.	In accordance with the INSERVICE TESTING PROGRAM

INSERT

1205 + 36.1, - 60.2
1195 + 35.8, - 59.7
1185 + 35.5, - 59.2
1175 + 35.2, - 58.7
1150 + 34.5, - 57.5

ATTACHMENT 4

Markup of Affected TS Bases Pages
(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 3.4.4-3
TS Bases Page B 3.4.4-4

BASES

LCO (continued)	The S/RV safety setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in Reference 3 involving the safety mode are based on these setpoints, but also include the additional uncertainties of ~~± 3%~~ +3% and -5% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.
	Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.
	The S/RVs are required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that MCPR is not exceeded.
APPLICABILITY	In MODES 1, 2, and 3, the specified number of S/RVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to limit peak reactor pressure.
	In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.
ACTIONS	<u>A.1 and A.2</u>
	With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If one or more required S/RVs are inoperable, the plant must be brought to

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance demonstrates that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the S/RV safety function lift settings must be performed during shutdown, since this is a bench test, and in accordance with the INSERVICE TESTING PROGRAM. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ~~± 3%~~ +3% and -5% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. Additionally, during the performance of this Surveillance, the S/RV will be manually actuated by providing air to the valve actuator to verify the performance of the valve actuator, lever and pivot mechanism to open the valve. A Note is provided to allow up to two of the required 12 S/RVs to be physically replaced with S/RVs with lower setpoints. This provides operational flexibility which maintains the assumptions in the overpressure protection analysis.

The Frequency is specified in the INSERVICE TESTING PROGRAM which requires the valves be subjected to a bench test during refueling outages. The Frequency is acceptable based on industry standards and operating history.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Section 5.2.2.1.3.
3. UFSAR, Chapter 15.
4. ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).
