November 17, 2017

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

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2, RELIEF FROM THE REQUIREMENTS OF THE ASME CODE AND OM CODE RE: RELIEF REQUESTS I4R-02, I4R-03, I4R-06, I4R-07, AND I4R-09, PROPOSED ALTERNATIVES TO VARIOUS INSERVICE INSPECTION INTERVAL (ISI) REQUIREMENTS OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME CODE), SECTION XI, 2007 EDITION WITH THE 2008 ADDENDA FOR THE FOURTH 10-YEAR ISI INTERVAL (EPID NOS. L-2017-LLR-0038 (CAC NOS. MF9760 AND MF9761), L-LR-2017-0076 (CAC NOS. MF9762 AND MF9763), L-2017-LLR-0033 (CAC NOS. MF9766 AND MF9767), L-2017-LLR-0035 (CAC NOS. MF9770 AND MF9771), AND L-2017-LLR-0037 (CAC NOS. MF9768, AND MF9769))

Dear Mr. Hanson:

By letter dated May 30, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17150A449), Exelon Generation Company, LLC (EGC, the licensee) submitted relief requests associated with the fourth Inservice Inspection (ISI) interval for LaSalle County Station (LSCS), Units 1 and 2. The fourth interval of the LSCS ISI Program is currently scheduled to begin on October 1, 2017, and end on September 30, 2027, and will comply with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2007 Edition with the 2008 Addenda.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(z)(1), the licensee requested the following relief requests for the LSCS fourth 10-year ISI interval:

- Relief Request I4R-02. Requests approval of alternative use of Boiling Water Reactor Vessel Intervals Project guidelines in lieu of specific ASME Code, Section XI requirements on the reactor pressure vessel (RPV) internals and components inspection.
- Relief Request I4R-03. Requests relief regarding examination of the RPV stabilizer bracket welds on shell course due to impracticality. By letter dated July 17, 2017 (ADAMS Accession No. ML17200C937), the licensee withdrew the request from the U.S. Nuclear Regulatory Commission (NRC).
- Relief Request I4R-06. Requests approval of alternative continuous pressure monitoring of the control rod drive system accumulators.

- Relief Request I4R-07. Requests approval of alternative pressure testing of the safety relief valve automatic depressurization system accumulators.
- Relief Request I4R-09. Requests alternative examination requirements for the nozzle-to-vessel welds and inner radii sections.

The other relief requests submitted by letter dated May 30, 2017 (ADAMS Accession No. ML17150A449), will be addressed via separate correspondence.

The NRC staff has reviewed the subject requests, as supplemented, and concludes, as set forth in the enclosed safety evaluations (SEs), that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). The NRC staff further concludes that the licensee is in compliance with the ASME Code requirements.

Therefore, the NRC authorizes the licensee's proposed alternatives from certain ISI requirements of the ASME Code as discussed in the SEs for the duration as described below:

(1) For I4R-02, -06, and -07: For the remainder of the fourth 10-year ISI interval for LSCS, Units 1 and 2.
(2) For I4R-09: For the fourth ISI Interval, as well as each 10-year ISI interval during the remaining term of the LSCS, Units 1 and 2, renewed facility operating licenses, which currently expires at midnight on April 17, 2042, and December 16, 2043, respectively. The Fourth ISI Interval, as well as the remaining term of the renewed facility operating licenses refers to the LSCS, Units 1 and 2 current fourth and upcoming fifth and sixth 120-month ISI Program intervals.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Please contact the Project Manager, Bhalchandra K. Vaidya at (301)415-3308, if you have any questions.

Sincerely,

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

Docket Nos. 50-373 and 50-374

Enclosures:
1. Enclosure 1 - Safety Evaluation – I4R-02, I4R-06, and I4R-07
2. Enclosure 2 - Safety Evaluation – I4R-09

cc: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I4R-02, I4R-06, AND I4R-07 FOR

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

LASALLE COUNTY STATION, UNITS 1 AND 2

EXELON GENERATION COMPANY, LLC

DOCKET NOS. 50-373 AND 50-374

RELIEF REQUEST I4R-02: ALTERNATIVE USE OF BOILING WATER REACTR VESSEL INTERNALS PROJECT (BWRVIP) GUIDELINES IN LIEU OF SPECIFIC ASME SECTION XI REQUIREMENTS ON THE REACTOR PRESSURE VESSEL (RVP) INTERNALS AND COMPONENTS INSPECTION.

RELIEF REQUEST I4R-06: ALTERNATIVE CONTINUOUS PRESSURE MONITORING OF THE CONTROL ROD DRIVE (CRD) SYSTEM ACCUMULATORS.

RELIEF REQUEST I4R-07: REQUESTS APPROVAL OF ALTERNATIVE PRESSURE TESTING OF THE SAFETY RELIEF VALVE (SRV) AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) ACCUMULATORS.

1.0 INTRODUCTION

By letter dated May 30, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17150A449), Exelon Generation Company, LLC (Exelon, the licensee), submitted requests for relief in I4R-02, I4R-06, and I4R-07 from certain inservice inspection (ISI) requirements of Section XI of the 2007 Edition through the 2008 Addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for the LaSalle County Station (LSCS), Units 1 and 2.

For I4R-02, the licensee requested relief from the requirements of the ASME Code for examination requirements associated with Table IWB-2500-1, Categories B-N-1 and B-N-2, and identified with Item Nos. B13.10, B13.20, B13.30, and B13.40. Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use BWRVIP guidelines in lieu of specific ASME Code, Section XI, requirements on the RPV internals and component inspections (i.e., vessel interior, interior attachments within beltline region, interior attachments beyond beltline region, and core support structure) on the basis that the alternative provides an acceptable level of quality and safety.

For I4R-06, the licensee requested relief from the requirements of the ASME Code for examination requirements associated with Table IWC-2500-1, Examination Category C-H, and identified with Item No. C7.10. Specifically, pursuant to 10 CFR 50.55a(z)(1), the licensee requested to use continuous pressure monitoring of the CRD system accumulators in lieu of the required system leakage test and VT-2 visual examination once each inspection period for all Class 2 pressure retaining components on the basis that the alternative provides an acceptable level of quality and safety.

For I4R-07, the licensee requested relief from the requirements of the ASME Code for examination requirements associated with Table IWC-2500-1, Examination Category C-H, and identified with Item No. C7.10. Specifically, pursuant to 10 CFR 50.55a(z)(1), the licensee requested to use alternative pressure testing of the SRV ADS accumulators in lieu of the required system leakage test and visual examination (VT)-2 once each inspection period for all Class 2 pressure retaining components on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY REQUIREMENTS

The ISI of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained. This is required by 10 CFR, Section 50.55a(g), except where specific relief has been granted by the U.S. Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 CFR 50.55a(g)(6)(i). The regulation in 10 CFR 50.55a(z) states that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The ASME Code of record for LSCS, Units 1 and 2, for the fourth 120-month ISI interval program, is the ASME Code, Section XI, 2007 Edition with the 2008 Addenda

3.0 EVALUATION

3.1 I4R-02 (LSCS, Units 1 and 2), Examination Categories B-N-1 and B-N-2, Code Item Nos. B13.10, "Vessel Interior," B13.20, "Interior Attachments within Beltline Region," B13.30, "Interior Attachments Beyond Beltline Region," and B13.40, "Core Support Structures."

3.1.1 The Licensee's Request for Alternative

Component(s) for which Alternative is Requested (ASME Code Class 1)

Vessel Interior, Interior Attachments within Beltline Region, Interior Attachments beyond Beltline Region, and Core Support Structure
Examination Category

B-N-1, and B-N-2, "Welded Core Support Structures and Interior Attachments to Reactor Vessels"

Examination Item Number

B13.10, "Vessel Interior"
B13.20, "Interior Attachments within Beltline Region"
B13.30, "Interior Attachments beyond Beltline Region"
B13.40, "Core Support Structures"

Applicable Code Edition and Addenda

ASME Code, Section XI, 2007 Edition with the 2008 Addenda

ASME Code Requirement for which Alternative is Requested

ASME Code, Section XI requires the visual examination (VT) of certain reactor vessel internal (RVI) components. These examinations are included in Table IWB-2500-1, Categories B-N-1 and B-N-2, and identified with the following item numbers:

- B13.10 - Examine accessible areas of the RVI interior each inspection period using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI

- B13.20 - Examine accessible interior attachment welds within the beltline region each interval using a technique which meets the requirements for a VT-1 examination as defined in paragraph IWA-2211 of the ASME Code, Section XI

- B13.30 - Examine accessible interior attachment welds beyond the beltline region each interval using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI

- B13.40 - Examine accessible surfaces of the core support structures each interval using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI

Licensee's Proposed Alternative to the ASME Code

In lieu of the requirements of ASME Code, Section XI, the licensee will satisfy the Examination Categories B-N-1 and B-N-2 requirements as described in Table 1 of its submittal in accordance with the latest NRC approved BWRVIP guideline requirements. This relief request proposes the use of the BWRVIP guidelines identified in the relief request in lieu of the associated ASME Section XI requirements, including examination method, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting. The licensee explained that the following guidelines are applicable to this relief request; however not all the components addressed by these guidelines are ASME Code, Section XI, components:

- BWRVIP-03, "BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines"
- BWRVIP-18, Revision 2-A, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines"
- BWRVIP-25, "BWR Core Plate Inspection and Flaw Evaluation Guidelines"
- BWRVIP-26-A, "BWR Top Guide Inspection and Flaw Evaluation Guidelines"
- BWRVIP-27-A, "BWR Standby Liquid Control System/Core Plate, ΔP Inspection and Flaw Evaluation Guidelines"
- BWRVIP-38, "BWR Shroud Support Inspection and Flaw Evaluation Guidelines"
- BWRVIP-41, Revision 3, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
- BWRVIP-42, Revision 1, "Low Pressure Coolant Injection (LPCI) Coupling Inspection and Flaw Evaluation Guidelines"
- BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
- BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
- BWRVIP-49-A, "Instrument Penetration Inspection and Flaw Evaluation Guidelines"
- BWRVIP-76, Revision 1-A, '"BWR Core Shroud Inspection and Flaw Evaluation Guidelines"
- BWRVIP-94NP, Revision 2, "Program Implementation Guide"
- BWRVIP-138, Revision 1-A "Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines"
- BWRVIP-180, "Access Hole Cover Inspection and Flaw Evaluation Guidelines"

The licensee stated that any deviations from the referenced BWRVIP guidelines for the duration of the proposed alternative will be appropriately documented and communicated to the NRC staff, per the BWRVIP deviation disposition process.

Licensee's Basis for Proposed Alternative

Boiling-water reactors (BWRs) examine reactor internals in accordance with BWRVIP guidelines. These guidelines were written to address the safety significant vessel internal components and to examine and evaluate the examination results for these components using appropriate methods and reexamination frequencies. The BWRVIP has established a reporting protocol for examination results and deviations.

The licensee provided with its submittal a summary of its previous reactor internals inspection history for LSCS, Units 1 and 2, which provides, on a component-by-component basis, the examination methods utilized, the examination frequency to date, the results of the examinations during the previous interval and the identified corrective actions (if applicable). Furthermore, the licensee also provided a comparison of ASME Code, Section XI, Examination Categories B-N-1 and B-N-2 requirements with BWRVIP guidance requirements, which provides specific examples comparing the inspection requirements of ASME Section X, Item Nos. B13.10, B13.20, B13.30, and B13.40 in Table IWB-2500-1, to the inspection requirements in the BWRVIP documents, including inspection methods.

The licensee explained that the reactor vessel internals (RVIs) inspection program at the site has been developed and implemented to satisfy the requirements of BWRVIP-94. It is recognized that the BWRVIP executive committee periodically revises the BWRVIP guidelines to include enhancements in inspection techniques and flaw evaluation methodologies. Where the revised version of a BWRVIP inspection guideline continues to also meet the requirements of the version of the BWRVIP inspection guideline that forms the safety basis for an NRC-authorized proposed alternative to the requirements of 10 CFR 50.55a, it may be implemented. Otherwise, the revised guidelines will only be implemented after NRC approval of the revised BWRVIP guidelines or a plant-specific request for relief has been approved.

3.1.2 <u>NRC Staff Evaluation</u>

The NRC staff found the referenced BWRVIP reports identified in the licensee's submittal (other than BWRVIP-180[1]) to be acceptable for use because the inspection and evaluation (I&E) guidelines addressed in these reports would identify degradation in a timely manner and ensures that the integrity of the RVI components will be maintained. In addition, for LSCS, Units 1 and 2, compliance with inspection criteria included in the reports would provide reasonable assurance that age related degradation in the RVIs components will be identified in a timely manner.

As part of its evaluation of this relief request, the NRC staff reviewed the results of the reactor internals inspection history up until the most recent inspections in 2016 and 2015 for LSCS, Units 1 and 2, respectively. These results were included as supporting information in the licensee's submittal and provide information regarding inspection methods used on the RVI components, inspection dates, the inspection results, and corrective actions related to the inspection findings.

Based on its review of these inspection summaries for LSCS, Units 1 and 2, the NRC staff finds that the licensee had adequately demonstrated its capability in: (1) identifying the weld flaws (cracking); (2) taking appropriate corrective actions to ensure that the structural integrity of the component is maintained (i.e., proper repair (if necessary), or flaw evaluation with proper engineering justification); and (3) complying with scope expansion of inspections and subsequent inspections per the applicable BWRVIP reports.

The NRC staff also finds that the inspection techniques that are recommended by the BWRVIP inspection guidelines meet or exceed the inspection techniques mandated by the ASME Code, Section XI, ISI program. In addition, the BWRVIP I&E guidelines provide inspection frequencies sufficient to detect aging degradation. Therefore, subsequent inspections of the RVI components per the relevant BWRVIP I&E guidelines will provide reasonable assurance that

---

[1] The BWRVIP-180 report was not submitted to the NRC for review and approval, however, the licensee can use this report provided the inspection guidelines that are recommended by this report meet or exceed the inspection techniques mandated by the ASME Code, Section XI, ISI pram.

any emerging aging effects will be identified in a timely manner. In addition, inspections per these guidelines will enable the licensee to effectively monitor the existing aging degradation in RPV interior surfaces, attachments, and core support structures. Based on the above considerations, the NRC staff finds that the implementation of the inspection requirements specified in the licensee's proposed alternative will ensure that the integrity of the RVI components will be maintained with an acceptable level of quality and safety.

The NRC staff acknowledges that the BWRVIP Executive Committee periodically revises the BWRVIP guidelines to include enhancements in inspection techniques and flaw evaluation methodologies. While the licensee may choose to implement enhancements described in a revised version of a BWRVIP inspection guideline, the licensee must continue to also meet the requirements of the version of the BWRVIP inspection guideline that forms the basis for the NRC staff's authorized alternative to the requirements of 10 CFR 50.55a. The licensee may, of course, also choose to return to complying with the inspection requirements of the ASME Code, code of record (COR) for LSCS, Units 1 and 2.

Thus, the NRC staff authorizes only the BWRVIP inspection guidelines proposed as an alternative in this relief request. In the event the licensee decides to take exceptions to, or deviations from, the authorized alternative, the licensee must revise and resubmit its request for authorization to use the proposed alternative under 10 CFR 50.55a.

3.2 <u>I4R-06 (LSCS, Units 1 and 2), Examination Category C-H, Item C7.10, All Pressure Retaining Components - Continuous Pressure Monitoring of the CRD System Accumulators and Associated Piping</u>

3.2.1 <u>The Licensee's Request for Alternative</u>

<u>Component(s) for which Alternative is Requested (ASME Code Class 1)</u>

CRD Accumulators and Associated Piping

<u>Examination Category</u>

C-H, "All Pressure Retaining Components"

<u>Examination Item Number</u>

C7.10, "Pressure Retaining Components"

<u>Applicable Code Edition and Addenda</u>

ASME Code, Section XI, 2007 Edition with the 2008 Addenda

<u>ASME Code Requirement for which Alternative is Requested</u>

Table IWC-2500-1, Examination Category C-H, Item Number C7.10, requires all Class 2 pressure retaining components be subject to a system leakage test be performed in accordance with IWC-5220 and subject to a VT-2 visual examination. This pressure test is to be conducted once each inspection period.

<u>Licensee's Proposed Alternative to the ASME Code</u>

As an alternative to the VT-2 visual examination requirements of Table IWC-2500-1, the licensee will perform continuous pressure decay monitoring for the nitrogen side of the CRD accumulators and associated piping and a weekly surveillance in accordance with technical specification surveillance requirement (TS SR) 3.1.5.1 that requires a physical walkdown of all CRD accumulators.

Relief is requested from the performance of system pressure tests and VT-2 visual examination requirements specified in Table IWC-2500-1 for the nitrogen side of the CRD system accumulators and associated piping on the basis that the requirements of TS SR 3.1.5.1 exceeds the code required examinations.

<u>Licensee's Basis for Proposed Alternative</u>

In accordance with 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

The LSCS, Units 1 and 2, TS SR 3.1.5.1 requires each control rod scram accumulator pressure to be equal to or greater than 940 pounds per square inch gauge (psig) for the control rod scram accumulator to be considered operable. The TS SR is required to be met whenever the unit is operating in Modes 1 and 2. The accumulator pressure is continuously monitored by system instrumentation and surveillance is performed on a weekly basis that requires a physical walkdown of all CRD accumulators. The walkdown is intended to identify any system air leaks and negative trending in system pressure. The accumulators are isolated from the source of makeup nitrogen, thus the continuous monitoring of the CRD accumulators currently functions as a pressure decay type test. The accumulators are maintained at a pressure of approximately 1100 psig during operation. Should accumulator pressure fall below 1000 psig (-15 psig), an alarm is received in the control room. The pressure drop for the associated accumulator is then recorded in the control room log, and the accumulator is recharged by station procedure LOP-RD-20, "Control Rod Accumulator Recharging/Water Removal." Other corrective actions, including soap bubble application to locate leakage or equipment repair are performed, as required, in accordance with the Corrective Action Program (CAP).

Since the monitoring of the nitrogen side of the accumulator at pressures consistent with the requirements of Table IWC-2500-1 is continuous, any degradation of the accumulator and associated piping would be detected by normal system instrumentation. The accumulators are normally passive components and are susceptible to slow developing failure modes. Corrosion and tubing connection integrity are the primary modes of failure. Continuous monitoring will detect degrading conditions of individual accumulators due to these failure modes before similar detection by the code required examination. The continuous monitoring of the CRD accumulators and associated piping exceeds the code requirement of inspecting the system once per inspection period. The additional VT-2 visual examination performed once per inspection period would not provide an increase in safety, system reliability, or structural integrity. In addition, performance of a VT-2 visual would require applying a leak detection solution to 185 accumulators per unit in an elevated dose rate area. This results in radiation exposure (estimated 108 millirem (mrem) three times per interval for a total of 324 mrem) without any added benefit in the level of quality and safety. This inspection would not be consistent with as low as reasonably achievable (ALARA) practices.

3.2.2 NRC Staff Evaluation

The ASME Code requires that a VT-2 visual examination be performed on all Class 2 pressure retaining components once during each inspection period. The VT-2 visual examination is a check for leakage of typically water-filled systems that must be performed at normal operating system pressure. However, the CRD system accumulators and associated piping are nitrogen-filled components that are maintained at a constant gas pressure in order to respond to a demand for CRD actuation. As such, a standard VT-2 visual examination would not provide an adequate means to identify potential leakage. In order to test these components for potential leakage, a soap bubble test applied to all surfaces of the subject components would be necessary.

The licensee's TS SR 3.1.5.1, for LSCS, Units 1 and 2, require each control rod scram accumulator pressure to be equal to or greater than 940 psig during Modes 1 and 2 for the control rod scram accumulator to be considered operable. Currently, the accumulator pressure is continuously monitored by system instrumentation and surveillance is performed on a weekly basis that requires a physical walkdown of all CRD accumulators. Should the accumulator pressure fall below an established threshold, an alarm is received in the control room. This pressure drop for the associated accumulator is then recorded in the control room log, and the accumulator is recharged by an existing station procedure as a corrective action. The licensee explained that other corrective actions, including soap bubble application to locate leakage or equipment repair are performed, as required, in accordance with its CAP.

During its review of relief request I4R-06, the staff noted that the licensee was previously approved for a similar alternative to use continuous pressure decay monitoring for the nitrogen side of the CRD accumulators in lieu of ASME Code requirements during the second and third 10-year ISI intervals. The staff's approval of the relief requests for the second and third 10-year ISI intervals are documented in its safety evaluations (SEs) dated June 28, 2002, (ADAMS Accession No. ML17150A449), and January 30, 2008 (ADAMS Accession No. ML073610587), respectively.

The NRC staff finds that continuous on-line monitoring of the CRD system accumulators: (1) exceeds the frequency of the ASME Code required examination and (2) will be capable of detecting a degraded condition prior to performing the required ASME Code examination during each inspection period. Thus, the staff determines that the licensee's alternative of continuous online monitoring of the CRD system accumulators provides an acceptable level of quality and safety.

3.3 I4R-07 (LSCS, Units 1 and 2), Examination Category C-H, Item C7.10, All Pressure Retaining Components - Alternative Pressure Testing of the Safety Relief Valve (SRV) Automatic Depressurization System (ADS) Accumulators and Associated Piping

3.3.1 The Licensee's Request for Alternative

Component(s) for which Alternative is Requested (ASME Code Class 2)

SRV ADS Accumulators and Associated Piping

Examination Category

C-H, "All Pressure Retaining Components"

Examination Item No.

C7.10, "Pressure Retaining Components"

Applicable Code Edition and Addenda

ASME Code, Section XI, 2007 Edition with the 2008 Addenda.

ASME Code Requirement for which Alternative is Requested

Table IWC-2500-1, Examination Category C-H, Item No. C7.10, requires all Class 2 pressure-retaining components be subject to a system leakage test be performed in accordance with IWC-5220 and subject to a VT-2 visual examination. This pressure test is to be conducted once each inspection period.

Licensee's Proposed Alternative to the ASME Code

As an alternative to the VT-2 visual examination requirements of Table IWC-2500-1, the licensee will perform pressure decay testing on the ADS accumulators and associated piping every refueling outage in accordance with surveillance procedure LOS-MS-R7, "Main Steam Safety Relief Valve Operability," for Units 1 and 2. Relief is requested from the performance of system pressure tests and the VT-2 visual examination requirements specified in Table IWC-2500-1 for the SRV ADS accumulators and associated piping on the basis that the existing LSCS surveillances provide an acceptable level of quality and safety.

Licensee's Basis for Proposed Alternative

In accordance with 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative will provide an acceptable level of quality and safety.

The LSCS operating surveillance LOS-MS-R7, performs operability testing of the main steam safety relief valves including the seven relief valves and accumulators per unit that are required to provide automatic depressurization. These surveillances are performed on a refueling outage frequency as a requirement of LSCS inservice testing program. One specific test that these surveillances perform is a pressure decay test of the ADS accumulators, associated piping and valves. The pressure decay test is performed by isolating and pressurizing the ADS accumulators and associated piping to the nominal operating pressure (i.e., 100 psig). The decay in pressure is then monitored through calibrated pressure measuring instrumentation. If the acceptable pressure decay criteria are exceeded, the surveillances identify appropriate troubleshooting steps to perform, including soap-bubble application to locate leakage. The pressure decay test performed as part of LOS-MS-R7 will identify any degradation of the ADS accumulators and associated piping. The volume tested by these surveillances encompasses the entire ASME Code, Section XI, boundary. These surveillances are performed on a greater frequency than the required period frequency of Table IWC-2500-1 and the test pressure is consistent with the pressure requirements of Table IWC- 2500-1. Thus, the testing performed during these surveillances will provide the same level of quality and safety as the pressure testing and the VT-2 visual examination requirements of Table IWC-2500-1. The additional VT-2 visual examination performed once per inspection period would not provide an increase in safety, system reliability, or structural integrity. In addition, performance of a VT-2 visual examination would require applying a leak detection solution to seven accumulators per unit and associated piping in an elevated dose rate area with limited access. This results in radiation exposure (estimated 1440 mrem three times per interval for a total of 4320 mrem) without any added benefit in the level of quality and safety. This inspection would not be consistent with ALARA practices.

3.3.2 NRC Staff Evaluation

The ASME Code requires that a VT-2 visual examination be performed on all Class 2 pressure retaining components once during each inspection period. The VT-2 visual examination is a check for leakage of typically water-filled systems that must be performed at normal operating system pressure. However, the ADS system accumulators and associated piping are compressed air filled components that are maintained at a constant pressure in order to respond to a demand for SRV actuation. As such, a standard VT-2 visual examination would not provide an adequate means to identify potential leakage. In order to test these components for potential leakage, a soap bubble test applied to all surfaces of the subject components would be necessary.

The licensee explained that an operating surveillance procedure requires operability testing of the main steam safety relief valves, including the seven relief valves and accumulators per unit, that are required to provide automatic depressurization. Of the several tests that are performed, one specific test is a pressure decay test of the ADS accumulators, and associated piping and valves. This test is performed by isolating and pressurizing the ADS accumulators and associated piping to the nominal operating pressure and the decay in pressure is then monitored through calibrated pressure measuring instrumentation. The licensee confirmed that the tests performed by the operating surveillance procedure encompasses the entire boundary required by the ASME Code, Section XI. The licensee explained that the pressure decay tests include corrective measures such as soap bubble testing to identify and locate the leakage, if the acceptable pressure decay criteria are exceeded for the ADS accumulators, and associated piping and valves.

During its review of relief request I4R-07, the NRC staff noted that the licensee was previously approved for a similar alternative to use pressure decay testing on the ADS accumulators in lieu of ASME Code requirements during the second and third 10-year ISI intervals. The staff's approval of the relief requests for the second and third 10-year ISI intervals are documented in its safety evaluations dated June 28, 2002, (ADAMS Accession No. ML17150A449), and January 30, 2008 (ADAMS Accession No. ML073610587), respectively.

The NRC staff finds that pressure decay tests performed on the main steam safety relief valve ADS accumulators and associated piping: (1) exceeds the frequency of the ASME Code required examination and (2) will be capable of detecting a degraded condition prior to performing the required ASME Code examination during each inspection period. Thus, the staff determines that the licensee's alternative to perform pressure decay tests of the Main Steam safety relief valve ADS accumulators provides an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee-proposed alternatives, as described in relief requests I4R-02, I4R-06, and I4R-07, provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1), and is in compliance with the ASME Code's requirements. Therefore, the NRC staff authorizes the licensee-proposed alternatives, as described in relief requests I4R-02, I4R-06, and I4R-07, at LSCS, Units 1 and 2 for the remainder of the fourth 10-year ISI interval.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: O. Yee, NRR/EVIB

Date of issuance: November 17, 2017

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I4R-09

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

LASALLE COUNTY STATION, UNITS 1 AND 2

EXELON GENERATION COMPANY, LLC

DOCKET NOS. 50-373 AND 50-374

1.0 INTRODUCTION

By letter dated May 30, 2017, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17150A449), Exelon Generation Company, LLC (the licensee) submitted relief request I4R-09 for the fourth inservice inspection (ISI) interval, as well as the remaining term of the LaSalle County Station (LSCS), Units 1 and 2, renewed facility operating licenses. The renewed licenses currently expire at midnight on April 17, 2042, and December 16, 2043, respectively.

The licensee requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI for examination requirements associated with Class 1 nozzle-to-vessel weld and nozzle inner radii, as delineated in Item No. B3.90, "Nozzle-to-Vessel Welds," and B3.100, "Nozzle Inside Radius Section," of Table IWB-2500-1, "Examination Category B-D, Full Penetration Welded Nozzles in Vessels - Inspection Program." Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative in Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds Section XI, Division 1," on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY REQUIREMENTS

The ISI of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained. This is required by 10 CFR 50.55a(g), except where specific relief has been granted by the U.S. Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(z) of 10 CFR states that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that: (1) the proposed alternative would provide an acceptable level of quality and safety or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

For all reactor pressure vessel (RPV) nozzle-to-vessel shell welds and nozzle inner radii, ASME Code, Section XI, requires 100 percent inspection during each 10-year ISI interval. However, Code Case N-702 provides an alternative, which reduces the inspection of RPV nozzle-to-vessel shell welds and nozzle inner radii areas from 100 percent to 25 percent of the nozzles for each nozzle type during each 10-year interval. This Code Case was conditionally approved in Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, Section XI, Division 1," dated August 2014. For application of ASME Code Case N-702, the licensee is required to address the conditions specified in RG 1.147, Revision 17 for ASME Code Case N-702:

> The applicability of Code Case N-702 must be shown by demonstrating that the criteria in Section 5.0 of NRC Safety Evaluation [SE] regarding BWRVIP [Boiling Water Reactor Vessel Internals Project]-108 dated December 19, 2007 (ML073600374) or Section 5.0 of NRC Safety Evaluation regarding BWRVIP-241 dated April 19, 2013 (ML13071A240) are met. The evaluation demonstrating the applicability of the Code Case shall be reviewed and approved by the NRC prior to the application of the Code Case.

BWRVIP-108, "Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radii" and BWRVIP-241, "Probabilistic Fracture Mechanics [PFM] Evaluation for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii" contain PFM analysis results supporting Code Case N-702. Both reports are for 40 years of operation. BWRVIP-241 contains additional PFM results supporting revision of the evaluation criteria under "Conditions and Limitations" in the SE for BWRVIP-108. The SE for BWRVIP-241 dated April 19, 2013, accepted the revised criteria.

Recently, the NRC issued an SE dated April 26, 2017, (ADAMS Accession No. ML17114A096) on a supplemental document for license renewal, BWRVIP-241, Appendix A, "BWR Nozzle Radii and Nozzle-to-Vessel Welds Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule (10 CFR 54.21)." This license renewal Appendix A extends the applicability of the BWRVIP-108 and BWRVIP-241 methodologies, and, therefore, Code Case N-702, from 40 years to the period of extended operation.

3.0 STAFF EVALUATION

3.1 The Licensee's Request for Alternative

Component(s) for which Alternative is Requested (ASME Code Class 1)

Reactor Pressure Vessel Nozzles, Units 1 and 2: N1, N2, N3, N5, N6, N7, N8, N9, N16, and N18

Examination Category

B-D, "Full Penetration Welded Nozzles in Vessels"

Examination Item Number

B3.90, "Nozzle-to-Vessel Welds"
B3.100, "Nozzle inside Radius Section"

Applicable Code Edition and Addenda

The fourth 10-Year ISI program at LSCS, Units 1 and 2, is based on the ASME Code, Section XI, 2007 Edition with the 2008 Addenda. Additionally, for ultrasonic examinations, ASME Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of the 2007 Edition with the 2008 Addenda is implemented, as required and modified by 10 CFR 50.55a(b)(2)(xv).

ASME Code Requirement for which Alternative is requested

The applicable requirements are contained in Table IWB-2500-1, "Examination Category B-D, Full Penetration Welded Nozzles in Vessels." Class 1 Reactor Vessel nozzle-to-vessel weld and nozzle inner radii examination requirements are delineated in Item No. B3.90, "Nozzle-to-Vessel Welds," and B3.100, "Nozzle Inside Radius Section." The required method of examination is volumetric. All nozzles with full penetration welds to the reactor vessel shell (or head) and integrally cast nozzles are examined each interval. All of the nozzle assemblies identified in the licensee's submittal are full penetration welds.

Licensee's Proposed Alternative to the ASME Code

In accordance with 10 CFR 50.55a(z)(1), relief is requested from performing the required examinations on 100 percent of the nozzle assemblies as identified in its submittal. As an alternative for all welds and inner radii identified in its submittal, the licensee proposes to examine a minimum of 25 percent of the LSCS, Units 1 and 2, nozzle-to-vessel welds and inner radii sections, including at least one nozzle from each system and nominal pipe size, in accordance with ASME Code Case N-702. For the nozzle assemblies identified in its submittal, this would mean 25 percent from each of the groups identified in Tables 5-1 and 5-2 of its submittal during each 120-month interval. BWRVIP-108 and BWRVIP-241 are referenced by the licensee as the technical basis for its use of Code Case N-702. The licensee stated that the proposed alternative provides an acceptable level of quality and safety based on the technical content of BWRVIP-108 and BWRVIP-241, as endorsed by the NRC SEs for these two BWRVIP reports.

Code Case N-702 stipulates that a VT-1 visual examination may be used in lieu of the volumetric examination for the inner radii (i.e., Item No. B3.100, "Nozzle Inside Radius Section"). ASME Code Case N-648-1 may be used with associated RG 1.147 conditions for the nozzles selected for examination. Volumetric examinations of the inside radius section of those reactor vessel nozzles selected for examination will be completed if ASME Code Case N-648-1 is not applied.

Licensee's Basis for Proposed Alternative

In its submittal, the licensee evaluated the five criteria identified in Section 5.0 of the NRC SE for BWRVIP-241. Based upon the licensee's evaluation, all LSCS RPV nozzle-to-vessel shell or head full penetration welds and nozzle inner radii sections, with the exception of the recirculation outlet nozzles (i.e., N1) on Unit 2, meet the general and nozzle-specific criteria in BWRVIP-241. BWRVIP-241, Section 6.0, notes that for plants having recirculation outlet nozzles with Condition 4 greater than 1.15, a plant-specific analysis following the approach described in this report may be able to justify values greater than 1.15.

Since the Unit 2, N1 nozzles did not meet the BWRVIP-241 criteria; the licensee performed a plant-specific analysis to qualify all the Unit 1 and Unit 2, nozzles by determining the probability of failure (PoF) based on operation for 60 years and assuming no inspections were performed in the initial 40 years of operation.

To address the elevated fluence issue of certain nozzles in the belt-line region of the reactor vessel, the fluence associated with the Unit 2 N6 nozzle (Low Pressure Coolant Injection (LPCI) nozzle) at the end of 60 years of operation was used as an input, since this nozzle has the highest fluence of all the Unit 1 and Unit 2 RPV nozzles. Furthermore, the bounding load cases were analyzed (i.e., Unit pressure, turbine generator trip-SCRAM and loss of feedwater pumps/isolation valves close transients) and the number of thermal cycles used in the analysis was based on the LSCS RPV thermal cycle diagrams.

The licensee stated that its evaluation of the bounding nozzle for Units 1 and 2 meet the NRC safety goal of 5E-6 per year; thus, the application of ASME Code Case N-702 to all the Unit 1 and Unit 2 nozzles listed in its submittal is acceptable.

3.2 NRC Staff Evaluation

3.2.1 BWRVIP-108, BWRVIP-241, and NRC Requirements

The NRC staff's SE for the BWRVIP-241 report specified plant-specific criteria that must be met for applicants proposing to use this alternative in Code Case N-702.

The BWRVIP-241 NRC SE, Section 5.0, "Conditions and Limitations," states that each licensee who plans to request relief from ASME Code, Section IX, requirements for RPV nozzle-to-vessel shell welds and nozzle inner radii sections may reference the BWRVIP-241 report as the technical basis for the use of Code Case N-702 as an alternative. However, each licensee should demonstrate the plant-specific applicability of the BWRVIP-241 report to its plant in the relief request by demonstrating that the general and nozzle-specific criteria are satisfied.

BWRVIP-241 documents additional PFM results supporting revision of the five evaluation criteria in BWRVIP-108. Since the objective of BWRVIP-241 is limited (i.e., revision of the limitations and conditions specified in the SE for the BWRVIP-108 report) it is considered as a supplement to BWRVIP-108 and not a replacement. Applicants requesting relief from the ASME Code, Section XI, inspection requirements on the subject RPV nozzles for their plants must demonstrate that the five plant-specific criteria are satisfied, so that BWRVIP-241 report results apply to their plants.

In the NRC staff's SE for BWRVIP-108, it was established that only the recirculation inlet and outlet nozzles need to be assessed since the conditional PoFs for other nozzles are an order of magnitude lower. It was also established there that only the driving force needs to be assessed since the nozzle material fracture toughness-related RT_{NDT} values used in the PFM analyses were based on data from the entire fleet of BWR RPVs, making the PFM analyses bounding with respect to fracture resistance.

Based on the above, BWRVIP-241 documents additional PFM analyses on the recirculation inlet and outlet nozzles having the highest driving force among the BWR fleet to demonstrate that the associated vessel PoF during normal operation (the limiting condition) is still consistent with the NRC safety goal; thus supporting the proposed revision of the five evaluation criteria. The NRC staff SE for BWRVIP-241 accepted the proposed revision of the five evaluation criteria in BWRVIP-108.

3.2.2 NRC Staff Evaluation of the Licensee Submissions

The licensee provided in its submittal dated May 30, 2017, its evaluation of the five criteria, including the driving force factors, or ratios, using plant-specific RPV and nozzle data, and compared them against the criteria established in the BWRVIP-241 SE dated April 19, 2013 for LSCS, Units 1 and 2. The NRC staff's review and evaluation of each criterion are document below:

Criterion 1

The licensee confirmed that TSs 3.4.11 for Units 1 and 2, limits the heatup and cooldown rates to ≤ 100 °F (degree Fahrenheit) in any 1-hour period. This heatup/cooldown rate is also described in the updated final safety analysis report, Section 5.2.3.3.1.7. Thus, the NRC staff finds that Criterion 1 established in the BWRVIP-241 SE, which limits the maximum RPV heatup/cooldown rate to less than 115°F/hour, is satisfied for LSCS, Units 1 and 2.

Criteria 2 and 3 - Recirculation inlet nozzles (N2)

The calculation of Criteria 2 and 3 for the N2 nozzle resulted in a maximum value of 1.064 and 1.134, respectively; thus, the staff finds that Criterion 2 (1.15) and Criterion 3 (1.47) established in the BWRVIP-241 SE is satisfied for LSCS, Units 1 and 2.

Criteria 4 and 5 - Recirculation outlet nozzles (N1)

The calculation of Criterion 4 for the N1 nozzle resulted in a maximum value of 1.025 for Unit 1 and 1.272 for Unit 2; thus, the NRC staff finds Criterion 4 (1.15) established in the BWRVIP-241 SE is satisfied for Unit 1, only. The calculation of Criterion 5 for the N1 nozzle resulted in a maximum value of 1.114; thus, the staff finds Criterion 5 (1.59) established in the BWRVIP-241 SE is satisfied for LSCS, Units 1 and 2.

With respect to the Unit 2, N1 nozzle, the licensee indicated that consistent with Section 6.0 of BWRVIP-241, plants having recirculation outlet nozzles with Condition 4 greater than 1.15 may perform a plant-specific analysis following the approach described in this BWRVIP-241 to justify values greater than 1.15.

The licensee explained that a plant-specific analysis was performed to qualify all the Unit 1 and Unit 2 nozzles and that the PoF was calculated based on operation for 60 years and assumes no inspections were performed in the initial 40 years of operation. In order for this plant-specific analysis to be bounding for LSCS, Units 1 and 2, the licensee accounted for the plant-specific geometry of the N1 nozzle (i.e., bounding nozzle for Units 1 and 2), addressed the elevated fluence levels due to 60 years of operation and incorporated 60-year thermal cycles counts. The licensee determined that for the nozzle blend radii and nozzle-to-shell weld, the PoF per year with 25 percent ISI through 60 years of plant operation (zero inspection during initial 40-year license) meets the NRC safety goal of 5E-6 per year for both the low temperature overpressure events and the normal operating condition.

The licensee accounted for the elevated fluence levels due to 60 years of operation by using the beltline nozzle with the highest fluence of all the Unit 1 or Unit 2 nozzles as an input to its analysis, which is the Unit 2 N6 nozzle (LPCI nozzle). The NRC staff noted that the analyses that supported BWRVIP-108 and BWRVIP-241 did not account for fluence since they only accounted for the original 40-year license of a plant. Thus, for 60 years of operation, the staff finds it appropriate that the licensee accounted for fluence levels at the end of 60 years of operation in its plant-specific PFM analysis. Furthermore, based on the fluence levels at the Unit 2, N6 nozzle, the NRC staff finds it acceptable that the fluence levels for Unit 2 N6 nozzle was used as a binding value for all the Unit 1 or Unit 2 RPV nozzles associated with its submittal.

The licensee indicated that only the bounding load cases were analyzed, which included the "Unit pressure," "Turbine Generator Trip-SCRAM," and "Loss of Feedwater Pumps/Isolation Valves Close" thermal transients applicable to the LSCS recirculation outlet nozzle. The staff finds this approach is consistent with PFM analyses supporting BWRVIP-241, which accounted for the bounding thermal transients applicable to two recirculation outlet nozzles selected in the report. The staff noted that the analyses that supported BWRVIP-108 and BWRVIP-241 only accounted for thermal transients for 40 years of operation; thus, the NRC staff finds it appropriate that the licensee accounted for thermal transients for 60 years of operation in its plant-specific PFM analysis.

Based on the licensee's plant-specific PFM analysis, which accounted for nozzle geometry, fluence levels and thermal transient cycles at the end of 60 years of operation, the staff determined that the reduced inspection requirements in Code Case N-702 apply to all proposed LSCS, Units 1 and 2, RPV nozzles (see Section 3.1 of this SE) for the fourth ISI interval and the remaining term of the LSCS, Units 1 and 2, renewed facility operating licenses. The proposed alternative also provides an acceptable level of quality and safety because the plant-specific PFM results meet the NRC safety goal on PoF through fourth ISI Interval and the remaining term of the LSCS, Units 1 and 2, renewed facility-operating licenses.

4.0 CONCLUSION

The NRC staff reviewed the licensee's evaluation of the five criteria specified in the NRC staff SE for the BWRVIP-241 report, which provides technical bases for Code Case N-702, to examine RPV nozzle-to-vessel welds and nozzle inner radii at LSCS, Units 1 and 2. As set forth above, the NRC staff determines that the licensee's proposed alternative provides an acceptable level of quality and safety and applies to all requested LSCS, Unit 1 and 2, RPV nozzles. However, this relief request does not include feedwater nozzles and control rod drive return nozzles. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) and is in compliance with the ASME Code's requirements. Therefore, the NRC authorizes the licensee's proposed alternative to use Code Case N-702 for inspection of nozzle-to-vessel shell welds and nozzle inner radii sections of RPV nozzles listed in Section 3.0 of this SE for the fourth ISI

Interval, as well as each 10-year ISI interval during the remaining term of the LSCS, Units 1 and 2, renewed facility operating licenses, which currently expires at midnight on April 17, 2042, and December 16, 2043, respectively. The Fourth ISi Interval, as well as the remaining term of the renewed facility operating licenses refers to the LSCS, Units 1 and 2 current fourth and upcoming fifth and sixth 120-month ISi Program intervals.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: O. Yee, NRR/EVIB

Date of issuance: November 17, 2017

SUBJECT: LASALLE COUNTY STATION, UNITS 1 AND 2, RELIEF FROM THE REQUIREMENTS OF THE ASME CODE AND OM CODE RE: RELIEF REQUESTS I4R-02, I4R-03, I4R-06, I4R-07, AND I4R-09, PROPOSED ALTERNATIVES TO VARIOUS INSERVICE INSPECTION INTERVAL (ISI) REQUIREMENTS OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME CODE), SECTION XI, 2007 EDITION WITH THE 2008 ADDENDA FOR THE FOURTH 10-YEAR ISI INTERVAL (EPID NOS. L-2017-LLR-0038 (CAC NOS. MF9760 AND MF9761), L-LR-2017-0076 (CAC NOS. MF9762 AND MF9763), L-2017-LLR-0033 (CAC NOS. MF9766 AND MF9767), L-2017-LLR-0035 (CAC NOS. MF9770 AND MF9771), AND L-2017-LLR-0037 (CAC NOS. MF9768, AND MF9769)) DATED NOVEMBER 17, 2017

DISTRIBUTION:
PUBLIC
PM Reading File
RidsNrrPMLaSalle Resource
RidsOgcRp Resource
RidsNrrDmlrMvib Resource
RidsRgn3MailCenter Resource
RidsNrrDorlLpl3 Resource
RidsNrrLASRohrer Resource
RidsNrrDorlDpr Resource
RidsAcrs_MailCTR Resource
OYee, NRR/MVIB

ADAMS Package Accession No. ML17305B279

(*) No Substantial change from SE Input Memorandum

OFFICE	NRR/LPL3/PM	NRR/LPL3/LA	NRR/DMLR/MVIB/BC(*)	NRR/LPL3/BC
NAME	BVaidya	SRohrer	DRudland/SRuffin	DWrona
DATE	11/17/17	11/02/17	08/18/17 & 09/14/17	11/17/17

OFFICIAL RECORD COPY