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

November 21, 2016

Mr. Bryan C. Hanson
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: SAFETY EVALUATION OF RELIEF REQUESTS I4R-02 AND I4R-10 FOR THE FOURTH 10-YEAR INTERVAL OF THE INSERVICE INSPECTION PROGRAM FOR LIMERICK GENERATING STATION, UNITS 1 AND 2 (CAC NOS. MF7587 AND MF7588)

Dear Mr. Hanson:

By letter dated April 13, 2016, as supplemented by letters dated May 11, 2016; July 12, 2016; and September 19, 2016, Exelon Generation Company, LLC submitted Relief Requests I4R-01, I4R-02, I4R-05, I4R-06, I4R-07, I4R-08, I4R-09, I4R-10, I4R-11, I4R-12, and I4R-13, which proposed alternatives to certain requirements specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) for the Limerick Generating Station (LGS), Units 1 and 2. The subject relief requests are for the fourth 10-year interval of the inservice inspection (ISI) program at LGS.

The purpose of this letter is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's review of Relief Requests I4R-02 and I4R-10, as documented in the enclosed safety evaluation (SE). Our SE concludes the following.

(1) With respect to Relief Request I4R-02, the proposed alternative will provide an acceptable level of quality and safety. Therefore, pursuant to Section 50.55a(z)(1) of Title 10 of the *Code of Federal Regulations* (10 CFR), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at LGS.

(2) With respect to Relief Request I4R-10, the proposed alternative will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(z)(1), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at LGS.

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.

The NRC staff will provide separate correspondence regarding the review for Relief Requests I4R-01, I4R-05, I4R-06, I4R-07, I4R-08, I4R-09, I4R-11, I4R-12, and I4R-13.

If you have any questions concerning this matter, please contact the LGS Project Manager, Mr. Richard Ennis, at (301) 415-1420 or Rick.Ennis@nrc.gov.

Sincerely,

Stephen S. Koenick, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosure:
Safety Evaluation

cc w/enclosure: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUESTS I4R-02 AND I4R-10 FOR THE

FOURTH 10-YEAR INTERVAL OF THE INSERVICE INSPECTION PROGRAM

EXELON GENERATION COMPANY, LLC

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated April 13, 2016, as supplemented by letters dated May 11, 2016; July 12, 2016; and September 19, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML16104A122, ML16132A441, ML16194A230, and ML16263A218, respectively), Exelon Generation Company, LLC (the licensee) submitted Relief Requests I4R01, I4R-02, I4R-05, I4R-06, I4R-07, I4R-08, I4R-09, I4R-10, I4R-11, I4R-12, and I4R-13, which proposed alternatives to certain requirements specified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) for the Limerick Generating Station (LGS), Units 1 and 2. The subject relief requests are for the fourth 10-year interval of the inservice inspection (ISI) program at LGS. The fourth 10-year ISI interval for LGS, Units 1 and 2, starts on February 1, 2017, and ends on January 31, 2027.

The purpose of this safety evaluation (SE) is to provide the results of the U.S. Nuclear Regulatory Commission (NRC) staff's review of Relief Requests I4R-02 and I4R-10. The NRC staff's review for Relief Requests I4R-01, I4R-05, I4R-06, I4R-07, I4R-08, I4R-09, I4R-11, I4R-12, and I4R-13, will be documented in separate SEs.

2.0 REGULATORY EVALUATION

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), the ISI of ASME Code Class 1, 2, and 3 components is performed in accordance with latest edition and addenda of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). Additionally, pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the Section XI of the ASME Code, to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulation requires that inservice examination of components and system pressure tests conducted during the first 10-year interval, and subsequent intervals, comply with the requirements of the latest edition and addenda of Section XI of the ASME Code, incorporated by reference in 10 CFR 50.55a(a), 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed in 10 CFR 50.55a(b). The applicable Code of Record for the fourth 10-year ISI interval for LGS, Units 1 and 2, is the ASME Code, Section XI, 2007 Edition through 2008 Addenda.

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (1) the proposed alternatives would provide an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

3.0 TECHNICAL EVALUATION

3.1 Relief Request I4R-02

3.1.1 Licensee's Request

Description of Relief Request

In Relief Request I4R-02, Revision 0, the licensee proposed use of Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines in lieu of specific ASME Code, Section XI requirements for reactor pressure vessel (RPV) internals and components inspection. The request was submitted pursuant to 10 CFR 50.55a(z)(1) on the basis that the proposed alternative would provide an acceptable level of quality and safety.

Components Affected

The relief request pertains to Section XI, ASME Code Class 1, 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 Structure).

Applicable Code Requirements

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

B13.10 Examine accessible areas of the RPV interior surfaces each 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 (B-N-1).

B13.20 Examine 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 (B-N-2).

B13.30 Examine 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 (B-N-2).

B13.40 Examine surfaces of the core support structure 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 (B-N-2).

These examinations are performed to assess the structural integrity of the RPV interior surfaces, attachments, and core support structures.

BWRVIP Guidelines

In the submittal dated April 13, 2016, the licensee, in lieu of ASME Code, Section XI requirements, submitted an alternative inspection program per the BWRVIP guidelines for Categories B-N-1 and B-N-2 RPV interior surfaces, attachments, and core support structures at LGS, Units 1 and 2. The licensee stated that implementation of the alternative inspection program will maintain an adequate level of quality and safety of the affected welds and components and will not adversely impact the health and safety of the public. The licensee stated that the owners of the boiling-water reactors (BWRs) in the United States now examine the RPV interior surfaces, attachments, and core support structures in accordance with BWRVIP guidelines. The proposed alternative covers examination methods, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting. The licensee stated that the BWRVIP guidelines have been written to address the examination of safety-significant RVI components using appropriate methods and re-inspections at conservative intervals. In contrast, the ASME Code re-inspection intervals are less conservative than the intervals stipulated in the BWRVIP inspection and evaluation (I&E) guidelines.

Licensee's Proposed Alternative

In lieu of the requirements of the applicable edition and addenda of Section XI of the ASME Code, the licensee proposed to examine the LGS, Units 1 and 2, RVI components in accordance with BWRVIP guidelines. In its request, the licensee included only the RVI components that come under the jurisdiction of Section XI of the ASME Code. The following reports include the relevant BWRVIP I&E guidelines for the RPV interior surfaces, attachments, and core support structures. Furthermore, the licensee clarified that not all RVI components listed in the following BWRVIP reports are ASME Code, Section XI components:

- BWRVIP-18, Revision 1-A, "BWRVIP Core Spray Internals Inspection and Flaw Evaluation Guidelines"
- BWRVIP-25, "BWRVIP Core Plate Inspection and Flaw Evaluation Guidelines"
- BWRVIP-26-A, "BWRVIP Top Guide Inspection and Flaw Evaluation Guidelines"
- BWRVIP-27-A, "BWRVIP BWR Standby Liquid Control System/Core Plate Delta P 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 System (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, "BWR Vessel Internals Project, Instrument Penetration Inspection and Flaw Evaluation Guidelines"
- BWRVIP-76, Revision 1-A, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"
- BWRVIP-94NP, Revision 2, "BWRVIP Program Implementation Guide"
- BWRVIP-138, Revision 1-A, "BWRVIP Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines"
- BWRVIP-139-A, "BWR Vessel Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines"
- BWRVIP-180, "Access Hole Cover Inspection and Flaw Evaluation Guidelines"
- BWRVIP-183-A, "BWRVIP, Top Guide Grid Beam Inspection and Flaw Evaluation"

The licensee stated that inspection services by an authorized inspection agency will be applied to the proposed alternative. The licensee further indicated that the BWRVIP has established reporting protocol for examination results and deviations that are consistent with the requirements of BWRVIP-94.

The licensee stated 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.

In Table 1 of Relief Request I4R-02, in the submittal dated April 13, 2016, the licensee provided a comparison of the ASME Code, Section XI examination requirements for Categories B-N-1 and B-N-2 for the RPV interior surfaces, attachments, and core support structures against the BWRVIP I&E guidelines examination requirements. In Attachment 1 to Relief Request I4R-02, the licensee provided additional information regarding the BWRVIP inspection requirements for the following welds (shown as examples) of the RPV interior surfaces, attachments, and core support structures and their subcomponents representing each of the ASME Code, Section XI, Item Nos. B13.10, B13.20, B13.30, and B13.40:

- Core Spray Piping (B13.10)
- Jet Pump (B13.20)
- Core Shroud (B13.30)
- Core Shroud Support and Core Support Structure (B13.40)

The licensee indicated that these examples demonstrate 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. Additionally, these examples demonstrated that the BWRVIP inspection guidelines require more frequent inspections of some RVI components compared to the corresponding ASME Code, Section XI, ISI program. In addition, the inspection techniques used in some of the BWRVIP reports have superior capability in detecting weld indications than the inspection techniques prescribed in the ASME Code, Section XI requirements. The licensee concluded that implementation of the BWRVIP inspection guidelines for the LGS, Units 1 and 2, RPV interior surfaces, attachments, and core

support structures, as an alternative to the subject ASME Code requirements, would provide an acceptable level of quality and safety.

The licensee proposed that the alternative be utilized through the remainder of the fourth ISI interval for LGS, Units 1 and 2.

3.1.2 NRC Staff Evaluation

The NRC staff found the referenced BWRVIP reports (other than BWRVIP-180[1]) to be acceptable for use because I&E guidelines addressed in these reports are consistent with ASME Code, Section XI, ISI criteria. In addition, for LGS, Units 1 and 2, compliance with inspection criteria included in the reports would provide reasonable assurance that the aging management program is effective in identifying the aging degradation in the RVI components in a timely manner.

As part of its evaluation of this relief request, the NRC staff reviewed the following Electric Power Research Institute (EPRI) documents:

- EPRI letter to the NRC dated April 21, 2015, "Project No. 704 – BWR Vessel and Internals Inspection Summaries for Spring 2014 Outages" (ADAMS Accession No. ML15117A061).
- EPRI letter to the NRC dated May 23, 2016, "Project No. 704 – BWR Vessel and Internals Inspection Summaries for Spring 2015 Outages" (ADAMS Accession No. ML16152A162).

The EPRI documents provided a compilation of information developed by individual utilities regarding RVI inspections for BWR plants. The letter dated April 21, 2015, included RVI inspection summaries for LGS, Unit 2. The letter dated May 23, 2016, included RVI inspection summaries for LGS, Unit 1. The letters provided information regarding inspection methods used on the RVI components, inspection dates, the results of the inspections, and corrective actions related to the findings of the inspections. The NRC staff reviewed the inspection summaries for LGS, Units 1 and 2, and concludes 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 I&E guidelines in the BWRVIP-18 (core spray internals) and BWRVIP-41 (jet pump assembly) reports contain provisions for the licensee to evaluate plant-specific leakage from both detected and postulated flaws. These provisions provide reasonable assurance that leakage will be bounded by the allowable leakage such that there is adequate core cooling consistent with the plant-specific loss-of-coolant accident analysis.

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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 program.

The NRC staff 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 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 concludes 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 of Record for LGS, Units 1 and 2.

The NRC staff authorizes only the BWRVIP inspection guidelines proposed as an alternative. 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.

3.1.3 NRC Staff Conclusion

Based on the above evaluation, the NRC staff concludes that the proposed alternative will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(z)(1), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at LGS.

All other requirements of ASME Code, Section XI for which the alternative has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

3.2 Relief Request I4R-10

3.2.1 Licensee's Request

Description of Relief Request

In Relief Request I4R-10, Revision 0, the license proposed alternative requirements for RPV nozzle-to-vessel weld and inner radii examinations, in lieu of specific ASME Code, Section XI requirements. The request was submitted pursuant to 10 CFR 50.55a(z)(1), on the basis that the proposed alternative would provide an acceptable level of quality and safety.

Components Affected

The relief request pertains to the following components:

ASME Code Class: 1

Reactor vessel nozzles: N2 (Recirculation Inlet Nozzles), N3 (Main Steam Nozzles), N5 (Core Spray Nozzles), N6 (Head Spray Nozzles), N7 (Head Vent Nozzle), N8 (Jet Pump Instrument Nozzles), and N17 (Residual Heat Removal Nozzles)

Examination Category: B-D (Full Penetration Welded Nozzles in Vessels)

Item Numbers: B 3.90 (Nozzle-to-Vessel Welds) and B3.100 (Nozzle Inside Radius Section)

Applicable Code Requirements

The licensee requested an alternative to the requirements of the ASME Code, Section XI, 2007 Edition with 2008 Addenda, Table IWB-2500-1, "Examination Category B-D, Full Penetration Welded Nozzle in Vessels." ASME Code Class 1 RPV nozzle-to-vessel weld and nozzle inner radii examination requirements are delineated in Table IWB-2500-1, Item Number B3.90, "Nozzle-to-Vessel Welds," and Item Number B3.100, "Nozzle Inside Radius Section," respectively. The required method of examination is volumetric. All nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles are examined each interval. All the nozzles associated with this relief request have full penetration welds. As such, 100 percent of the nozzle assemblies associated with this relief request are currently required to be inspected each 10-year ISI interval.

Licensee's Proposed Alternative

Pursuant to 10 CFR 50.55a(a)(z)(1), relief is requested from performing the required examinations on 100 percent of the nozzle assemblies identified in Tables 5-1 and 5-2 of the subject relief request as shown in the licensee's submittal dated April 13, 2016. As an alternative, for all welds and inner radii identified in Tables 5-1 and 5-2, the licensee proposes to examine a minimum of 25 percent of the nozzle-to-vessel welds and inner radius sections, including at least one nozzle from each system and nominal pipe size, in accordance with ASME Code Case N-702, "Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1." Code Case N-702 stipulates that a VT-1 examination may be used in lieu of the volumetric examination for the inner radii. However, the licensee stated that it will perform volumetric examinations of all nozzle inside radius sections selected per this relief request.

The licensee included the following information in the April 13, 2016, submittal regarding the basis for the proposed alternative:

Electric Power Research Institute (EPRI) Technical Report 1003557, "BWRVIP-108: Boiling Water Reactor Vessel and Internals Project Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," provides the basis for

ASME Code Case N-702. The evaluation found that failure probabilities at the nozzle blend radius region and nozzle-to-vessel shell weld due to a Low Temperature Overpressure event are very low (i.e., $<1 \times 10^{-6}$ for 40 years) with or without inservice inspection. The report concludes that inspection of 25 percent of each nozzle type is technically justified.

This EPRI report was approved by the NRC in a Safety Evaluation (SE) dated December 19, 2007 (i.e., ADAMS Accession No. ML073600374). Section 5.0, "Plant-Specific Applicability," of the SE indicates that each licensee who plans to request relief from ASME Code, Section XI requirements for RPV nozzle-to-vessel shell welds and nozzle inner radii sections may reference the BWRVIP-108 report as the technical basis for the use of ASME Code Case N-702 as an alternative. However, each licensee should demonstrate the plant specific applicability criteria from the BWRVIP-108 report to its units in the relief request by showing that all the general and nozzle-specific criteria [as described in Enclosure 1 to Relief Request I4R-10] are satisfied...

Enclosure 1 to Relief Request I4R-10, in the submittal dated April 13, 2016, provided the licensee's analysis of the BWRVIP-108 plant-specific applicability criteria for LGS, Units 1 and 2. The results of the analysis are as follows:

Criterion 1: The maximum RPV heatup/cooldown rate is less than 115 degrees Fahrenheit (°F)/hour

LGS, Units 1 and 2, Technical Specification (TS) 3.4.6, "Pressure/Temperature Limits," provides a limiting condition for operation (LCO) of 100 °F per hour. The heatup/cooldown rate is referenced in the plant operating procedures. This heatup/cooldown rate is also described in the LGS Updated Final Safety Analysis Report (UFSAR), Section 5.3.3.6, "Operating Conditions."

Criterion 2: For recirculation inlet nozzles (N2), $(pr/t)/C_{RPV} < 1.15$

$(pr/t)/C_{RPV} = 1.032 < 1.15$ (i.e., criterion met)

Where:
p = RPV normal operating pressure
r = RPV inner radius
t = RPV wall thickness
$C_{RPV} = 19332$

Criterion 3: For recirculation inlet nozzles (N2), $[p(r_o^2+r_i^2)/ (r_o^2-r_i^2)]/C_{NOZZLE} < 1.15$

$[p(r_o^2+r_i^2)/ (r_o^2-r_i^2)]/C_{NOZZLE} = 0.950 < 1.15$ (i.e., criterion met)

Where:
p = RPV normal operating pressure
r_o = nozzle outer radius
r_i = nozzle inner radius
C_{NOZZLE} = 1637

Criterion 4: For recirculation outlet nozzles (N1), $(pr/t)/C_{RPV} < 1.15$

$(pr/t)/C_{RPV} = 1.234 > 1.15$ (i.e., criterion not met)

Where:
p = RPV normal operating pressure
r = RPV inner radius
t = RPV wall thickness
C_{RPV} = 16171

Criterion 5: For recirculation outlet nozzles (N1), $[p(r_o^2+r_i^2)/(r_o^2-r_i^2)]/C_{NOZZLE} < 1.15$

$[p(r_o^2+r_i^2)/(r_o^2-r_i^2)]/C_{NOZZLE} = 1.023 < 1.15$ (i.e., criterion met)

Where:
p = RPV normal operating pressure
r_o = nozzle outer radius
r_i = nozzle inner radius
C_{NOZZLE} = 1977

Based on the above results, the licensee concluded that all LGS nozzle-to-vessel shell full penetration welds and nozzle inner radii sections, with the exception of the recirculation outlet nozzles, meet the general and nozzle-specific criteria in BWRVIP-108. Since the criterion was not met for the recirculation outlet nozzles, these nozzles were not included in the relief request. The license concluded that ASME Code Case N-702 is appplicable to the nozzles included in the relief request.

The licensee proposes that the alternative will be utilized through the remainder of the fourth ISI interval for LGS, Units 1 and 2.

3.2.2 NRC Staff Evaluation

The NRC staff's SE dated December 19, 2007 (ADAMS Accession No. ML073600374), for BWRVIP-108, specified five plant-specific criteria that licensees must meet in order to demonstrate that the BWRVIP-108 report results apply to their plants. The five criteria that are addressed in the staff's SE for BWRVIP-108 were developed using the probabilistic fracture mechanics (PFM) analysis for the recirculation inlet and outlet nozzles. The SE stated that the nozzle material fracture toughness-related RT_{NDT} (reference temperature for nil ductility transition) values used in the PFM analyses were based on data from the entire fleet of BWR RPVs. Therefore, the BWRVIP-108 report PFM analyses are bounding with respect to fracture resistance for the recirculation inlet and outlet nozzles. The SE also stated that except for the RPV heatup/cooldown rate, the plant-specific criteria are only applicable to the recirculation inlet

and outlet nozzles, because the probabilities of failure for other nozzles are an order of magnitude lower.

The licensee stated that Criterion 1 is satisfied because TS 3.4.6.1 for LGS, Units 1 and 2, provides an LCO specifying a maximum heatup or cooldown of 100 °F/hour in any 1-hour period. This heatup/cooldown rate requirement is also stated in Section 5.3.3.6, "Operating Conditions," of the UFSAR. Criterion 1, applies to normal operating conditions. TS 3.4.6.1 is applicable "at all times" (i.e., in every plant mode). Therefore, TS 3.4.6.1 provides reasonable assurance that Criterion 1 will be met for both LGS units.

For the remaining criteria, in Enclosure 1 to Relief Request I4R-10, in the submittal dated April 13, 2016, the licensee provided plant-specific data and its evaluation against the criteria established in the NRC staff's SE for BWRVIP-108. The licensee's calculated results show that three of the remaining four criteria are satisfied, with the exception being Criterion 4 (as shown above). Independent staff calculations confirmed the accuracy of the licensee results. Based on the results for Criterion 4, the licensee appropriately excluded the LGS, Units 1 and 2, recirculation outlet nozzles from the scope of this relief request. The staff finds the analysis for the remaining nozzles acceptable.

The licensee will be performing volumetric examination in lieu of visual examination of the inner radii of the nozzle during its scheduled ASME Code, Section XI (Table IWB-2500-1), ISI examinations. Based on this, the NRC staff concludes that any aging degradation in the inner radii of the nozzles would be detected in a timely manner.

In response to an NRC staff's request for additional information, in its letter dated July 12, 2016 (ADAMS Accession No. ML16194A230), the licensee provided a synopsis of previous inspections and whether any indications were found in the components for which the alternative was requested. The licensee indicated that no recordable indications had been found during third interval inspections of the subject nozzles. In addition, the licensee stated that during the weld inspection conducted during the third interval, in some welds, it obtained inspection coverage of 75 percent, and for some welds, the licensee obtained an inspection coverage of 100 percent. Based on review of this information, the staff concludes that there is reasonable assurance that there is currently no active aging degradation in the subject welds.

Based on the above evaluation, the NRC staff concludes that the proposed alternative would provide an acceptable level of quality and safety for the following reasons: (1) the licensee's calculated results satisfy acceptance criteria 1, 2, 3, and 5 in the staff's SE for BWRVIP-108; (2) the licensee will be performing volumetric examination of the inner radii of the nozzles during its scheduled ASME Code, Section XI, ISI examinations, which provide reasonable assurance of detecting cracking in the inner radii of the nozzles in a timely fashion; and (3) the previous examinations on the subject welds revealed no recordable indications; therefore, there is reasonable assurance that there is currently no active aging degradation in the subject welds.

3.2.3 NRC Staff Conclusion

Based on the above evaluation, the NRC staff concludes that the proposed alternative will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(z)(1), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at LGS.

All other requirements of ASME Code, Section XI for which an alternative has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: G. Cheruvenki
R. Ennis

Date: November 21, 2016

If you have any questions concerning this matter, please contact the LGS Project Manager, Mr. Richard Ennis, at (301) 415-1420 or Rick.Ennis@nrc.gov.

Sincerely,

/RA/

Stephen S. Koenick, Acting Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-352 and 50-353

Enclosure:
Safety Evaluation

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DATE	11/03/2016	11/03/2016	11/06/2016	11/21/2016	11/21/2016

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