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

May 30, 2018

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

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – RELIEF REQUESTS I5R-02, I5R-03, AND I5R-04 FOR ALTERNATIVES TO CERTAIN ASME CODE REQUIREMENTS (CAC NOS. MG0116, MG0117, AND MG0118; EPID L-2017-LLR-0083, EPID L-2017-LLR-0084, AND EPID L-2017-LLR-0085)

Dear Mr. Hanson:

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML17223A280 and ML17346A153, respectively), Exelon Generation Company, LLC (Exelon, the licensee) submitted three relief requests to the U.S. Nuclear Regulatory Commission (NRC) for relief from certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV Code), Section XI requirements at the James A. FitzPatrick Nuclear Power Plant. The relief requests are associated with the fifth inservice inspection interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternatives on the basis that the alternatives provide an acceptable level of quality and safety.

The NRC staff has reviewed the subject requests and concludes, as set forth in the enclosed safety evaluations, that Exelon has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1).

All other ASME Code requirements for which relief was not specifically requested and approved in the subject requests remain applicable.

If you have any questions, please contact Tanya Hood at 301-415-1387 or Tanya.Hood@nrc.gov.

Sincerely,

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-333

Enclosures:
1. Safety Evaluation for Relief Request I5R-02
2. Safety Evaluation for Relief Request I5R-03
3. Safety Evaluation for Relief Request I5R-04

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I5R-02, REVISION 0, TO ALLOW USE OF BOILING WATER REACTOR

VESSEL AND INTERNALS PROJECT GUIDELINES

EXELON GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML17223A280 and ML17346A153, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Request I5R-02, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for its fifth 10-year inservice inspection (ISI) interval regarding inspection of its reactor vessel internals (RVIs) components at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick). In this safety evaluation, the term "RVI components" includes reactor vessel (RV) interior surfaces, attachments, and core support structures.

Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed to use Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines as an alternative to certain requirements of Section XI, "Rules for ISI of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for ISI of RVI components. The regulation at 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY REQUIREMENTS

Section 50.55a(g) of 10 CFR states, in part, that ISI of certain ASME Code Class 1, 2, and 3 systems and components be performed in accordance with Section XI of the ASME Code and applicable addenda incorporated by reference in the regulations, as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC 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 would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Section 50.55a of 10 CFR

allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The regulations require that inservice examination of components and system pressure tests conducted during the successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"), subject to the conditions listed in 50.55a(b). The applicable ASME Code of record for the fifth 10-year ISI interval for FitzPatrick is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 LICENSEE'S EVALUATION

3.1 Applicable Code Requirements

The ASME Code, Section XI, requires a visual examination (VT) of certain 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 RV interior during each period using a VT-3 examination, as defined in paragraph IWA-2213 of Section XI of the ASME Code.

- B13.20 - Examine interior attachment welds within the RV beltline region during each interval using a VT-1 examination, as defined in paragraph IWA-2211 of Section XI of the ASME Code.

- B13.30 - Examine interior attachment welds outside of the beltline region during each interval using a VT-3 examination, as defined in paragraph IWA-2213 of Section XI of the ASME Code.

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

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

3.2 Components for Which Relief is Requested

The ASME Code, Section XI, Class 1, Examination Categories B-N-1 and B-N-2, Code Item Numbers B13.10, Vessel Interior; B13.20, Interior Attachments within Beltline Region; B13.30, Interior Attachments Beyond Beltline Region; and B13.40, Core Support Structure.

3.3 Licensee's Reason for Request

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. As part of its justification for the relief, the licensee stated that boiling-water reactors (BWRs) now examine the RV interior surfaces, attachments, and core support structures in accordance with BWRVIP inspection and evaluation (I&E) guidelines in lieu of ASME Code, Section XI criteria. The BWRVIP guidelines are written for the safety-significant RVI components and provide appropriate examination and evaluation criteria using appropriate methods and reexamination frequencies. The proposed alternative includes examination methods, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting. Furthermore, the licensee stated that this alternative to the ASME Code, Section XI requirements is requested pursuant to 10 CFR 50.55a(z)(1).

3.4 Licensee's Proposed Alternative and Basis for Use

In lieu of the requirements specified in Section XI of the ASME Code, the licensee proposed to examine the FitzPatrick RVI components in accordance with BWRVIP I&E guideline requirements in the following BWRVIP reports for RV surfaces, attachments, and core support structures.

- BWRVIP-03, "BWRVIP Reactor Pressure Vessel and Internals Examination Guidelines"
- 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, "BWRVIP Shroud Support Inspection and Flaw Evaluation Guidelines"
- BWRVIP-41, Revision 3, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
- BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
- BWRVIP-48-A, "Vessel ID [Inside Diameter] Attachment Weld Inspection and Flaw Evaluation Guidelines"
- BWRVIP-76, Revision 1-A, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"
- BWRVIP-138, Revision 1-A, "Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines"
- BWRVIP-180, "Access Hole Cover Inspection and Flaw Evaluation Guidelines"

- BWRVIP-183, "Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines"
- BWRVIP-94, Revision 2, "BWRVIP Program Implementation Guide"

The licensee stated that inspection services by an authorized inspection agency will be applied to the proposed alternative actions of this relief request. The licensee further indicated that results of examinations and deviations for the BWR fleet are reported under an established protocol between the BWRVIP and the NRC. Also, since the BWRVIP guidelines are revised periodically, the licensee clarified that if new guidance includes changes that are less conservative than those approved by the NRC, this less conservative guidance shall be implemented only after NRC approval.

The licensee provided a comparison of the ASME Code, Section XI examination requirements for Categories B-N-1 and B-N-2 for RV surfaces, attachments, and core support structures with the current BWRVIP guideline requirements, as applicable to FitzPatrick in Table 1 of its submittal. In Enclosure 1 of its submittal, the licensee provided additional justification regarding the comparison of the inspection requirements of the ASME Code, Section XI, Table IWB-2500-1, Item Numbers B13.10, B13.20, B13.30, and B13.40, to the inspection requirements in the BWRVIP guidance documents. For example, the following excerpt from Enclosure 1 of its submittal indicates the applicable ASME Code, Section XI category/item numbers that are applicable to some of the FitzPatrick RVI components:

- Core Spray Piping, Top Guide, Jet Pump Welds and Components, etc. – Item No. B13.10
- Jet Pump Riser Brace-to-RV Wall Pad Welds – Item No. B13.20
- Core Spray Piping Bracket Welds – Item No. B13.30
- Core Shroud – Item No. B13.40

Based on examination method, scope, frequency, and flaw evaluation criteria, the licensee stated that the above examples demonstrate that the inspection techniques recommended by the BWRVIP I&E guidelines are equivalent to, or superior to, the inspection techniques mandated by the ASME Code, Section XI ISI program. For instance, the BWRVIP's inspection of jet pump riser braces per BWRVIP-41 uses enhanced VT-1 (EVT-1), whereas the ASME Code uses VT-1. The BWRVIP's inspection of core spray piping bracket welds per BWRVIP-48-A uses EVT-1 every 8 years for plants with a 2-year fuel cycle, whereas the ASME Code uses VT-3 every 10 years.

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information in Relief Request I5R-02, Revision 0, including the supplemental information in the letter dated December 11, 2017. The NRC staff reviewed the status of the referenced BWRVIP reports and found application of the referenced BWRVIP reports to be acceptable, provided that the NRC conditions associated with the latest safety evaluation for each BWRVIP report are implemented. By e-mail dated November 29, 2017 (ADAMS Accession No. ML17335A100), the NRC staff issued a request for additional information (RAI) requesting the licensee to explain why the current basis for examining the core plate would not be sufficient to manage either stress relaxation or cracking of the core plate rim hold-down bolts during the period of extended operation. The RAI request also asked for clarification regarding inspection results for Item B13.10 and justification for why BWRVIP-139 is not needed for either steam dryer hold-down brackets or steam dryer support brackets. Following is the NRC staff's evaluation of the licensee's response.

4.1 Comparison of ASME Examination Category B-N-1 Requirements with BWRVIP Guidance Requirements

Except for the B13.10 component (RV interior), which belongs to Examination Category B-N-1, all other subject components in Table 1 of the licensee's submittal belong to Examination Category B-N-2. For the Category B-N-1 RV interior, it should be noted that portions of the various examinations required by the applicable BWRVIP guidelines require access to accessible areas of the RV during each refueling outage. Examination of core spray piping and spargers (BWRVIP-18, Revision 2-A), top guide (BWRVIP-26-A), jet pump welds and components (BWRVIP-41, Revision 3), interior attachments (BWRVIP-48-A), core shroud welds (BWRVIP-76, Revision 1-A), shroud support (BWRVIP-38), and lower plenum components (BWRVIP-47-A), all provide such access. Examining specific welds and components within the RV interior above and below the core and the surrounding annulus area using remote camera essentially performs equivalent VT-3 examination of many areas in the RV interior. This visual examination per BWRVIP reports is more frequent than that required by ASME Code, Section XI. The licensee further stated that evidence of wear; structural degradation; loose, missing, or displaced parts; foreign materials; and corrosion product buildup have been observed during the course of implementing these BWRVIP examination requirements.

Based on its review, the NRC staff finds that the specified BWRVIP guideline requirements meet the subject Code requirements for examination method and frequency of the RV interior. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

4.2 Comparison of ASME Examination Category B-N-2 Requirements with BWRVIP Guidance Requirements

Regarding the Table 1 comparison of the current ASME Code, Section XI examination requirements with the current BWRVIP guideline requirements for Category B-N-2 items of the licensee's submittal, the NRC staff noted that for Item B13.20, the proposed BWRVIP examination methods are EVT-1 for one component and VT-1 for another, as opposed to VT-1 specified in the ASME Code, Section XI, for both components. Similarly, for Items B13.30 and B13.40, the proposed BWRVIP examination methods for the majority of the components are EVT-1 or ultrasonic testing, as opposed to VT-3 specified in the ASME Code, Section XI. For the examination frequency, Table 1 in the submittal shows that except for one Item B13.20 component, which has a longer examination interval (12 years versus 10 years), all other B13.20, B13.30, and B13.40 components have equivalent or shorter examination intervals. For this single B13.20 component, the slightly longer examination interval is compensated for by the better EVT-1 examination method. Therefore, for both the examination methods and frequency, the BWRVIP guidelines are equivalent or exceed the ASME Code, Section XI requirements.

Table 1 information in the submittal regarding integrally welded core support structure under Item B13.40 requires further evaluation because BWRVIP-25 is not listed as "Applicable BWRVIP Document," and the core plate is not in the "BWRVIP Exam Scope." It should be noted that two approaches in managing the core plate integrity in BWRVIP-25 have been accepted in Section 3.0.3.2.7, "BWR Vessel Internal Programs," of the license renewal safety evaluation for FitzPatrick, "Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant" (ADAMS Accession No. ML080250372). They are (1) to install core plate wedges prior to the period of extended operation and (2) to complete a plant-specific analysis to determine acceptance criteria for continued inspection of the core plate rim hold-down bolting in accordance with BWRVIP-25. In the RAI e-mail dated November 29, 2017, the NRC staff asked the licensee in RAI-1 to confirm whether it needs to revise Table 1 by including BWRVIP-25 as one of the applicable BWRVIP documents for Item B13.40 so that it can perform either BWRVIP-25 option approved in the license renewal safety evaluation for FitzPatrick to manage the core plate integrity. In its RAI response dated December 11, 2017, the licensee added BWRVIP-25 to Table 1 and provided Relief Request I5R-02, Revision 1.

Based on its review, the NRC staff finds that the proposed use of BWRVIP-25 guidance is another example of exceeding the ASME Code, Section XI, Table IWB 2500-1, B-N-2 requirements. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

4.3 Operating Experience and Flaw Evaluation

4.3.1 *Reactor Internals Inspection History*

The NRC staff reviewed the information in Relief Request I5R-02, Revision 0, including the supplemental information in the December 11, 2017, submittal. The NRC staff reviewed the reactor internals inspection history to assess the impact of using the BWRVIP inspections and disposition of indications on FitzPatrick RVI integrity. The NRC staff found no inspection record for ASME Code Item B13.10, "Reactor Vessel Interior." In its RAI e-mail dated November 29, 2017, the NRC staff asked the licensee in RAI-2 to clarify whether the absence of inspection results for ASME Code Item B13.10 meant that no relevant indications were noted for this item in all past examinations. In its RAI response dated December 11, 2017, the licensee confirmed that a review of examination results since 2000 for Item B13.10 has not identified any indications that were rejected by the ASME Code, Section XI.

Based on its review, the NRC staff finds this response acceptable. Therefore, the NRC staff concludes that the proposed alternative provides an acceptable level of quality and safety.

4.3.2 *Flaw Evaluation Guidelines and Plant-Specific Leakage Assessment*

The licensee does not mention the evaluation criteria for B13.20 components (B-N-2) and for the B13.40 component (B-N-2) core shroud. Enclosure 1 to Relief Request I5R-02, Revision 0, states, without elaboration, that comparable flaw evaluation criteria were used. The NRC staff examined the part of BWRVIP-48-A relevant to the B13.20 components, and the part of BWRVIP-38 and BWRVIP-76, Revision 1-A, relevant to the B13.40 components, and confirmed that the evaluation criteria for them, although not identical to the ASME Code, Section XI, were accepted by the NRC staff based on technical equivalency. It should be noted that although BWRVIP-38 does not have an "A" affix, a final safety evaluation for it was issued on July 24, 2000 (ADAMS Accession No. ML003735498). Further, Enclosure 2 to Relief Request I5R-02 indicates that, historically, all indications in various RVI components were satisfactorily dispositioned by repair or evaluations, and all follow-up inspections showed no meaningful change.
To further address cracking in several B-N-1 and B-N-2 components at FitzPatrick, as shown in Enclosure 2 to Relief Request I5R-02, Revision 0, the licensee performed plant-specific leakage

assessments in accordance with BWRVIP requirements for identified or postulated through-wall cracking in (1) the core spray 190-degree downcomer weld, (2) jet pump diffuser welds, and (3) core shroud welds. Based on conservative assumptions, leakages were calculated to be 40 gallons per minute (gpm) through the core spray 190-degree downcomer crack repair, 98.5 gpm for the jet pump welds, and 205 gpm for all known and postulated core shroud cracking. They are within the allowable limits of 123 gpm for Case (1) and 200 gpm for Case (2). For Case (3), the licensee used shroud leakage as a direct input to the FitzPatrick loss-of-coolant accident analysis, and the results indicated that there is no increase to the peak cladding temperature. The licensee stated that this plant-specific peak cladding temperature is below the 10 CFR 50.46(b) regulatory limit of 2,200 degrees Fahrenheit (°F).

Flaw evaluations are not required for B-N-1 components because the purpose of the examination is not to detect flaws, but rather to identify conditions such as distortion or displacement of parts; loose, missing, or fractured fasteners; foreign material; corrosion; erosion; wear; and structural degradation. The flaw evaluation methodologies for various B-N-2 components in the referenced BWRVIP reports are either the ASME Code, Section XI methodologies or accepted by the NRC staff based on acceptable levels of quality and safety. Subsequent inspections of the RVI components at FitzPatrick using the relevant BWRVIP I&E guidelines will provide reasonable assurance that emerging aging effects will be identified in a timely manner because (1) the FitzPatrick RVI inspection program has been developed and implemented to meet the requirements of the relevant BWRVIP reports, and (2) the BWRVIP I&E guidelines require the same or more frequent inspections than ASME Code, Section XI criteria for RVI components that are susceptible to aging degradation mechanisms.

In addition, frequent inspections in accordance with the BWRVIP I&E guidelines will enable the licensee to effectively monitor existing aging degradation in RV surfaces, attachments, and core support structures during the fifth ISI interval. For the associated plant-specific leakage assessments, the NRC staff concludes that they are acceptable because the leakage through the conservatively postulated core shroud cracks, combined with leakage in jet pump welds and the core spray weld, would not increase the peak cladding temperature analyzed in the FitzPatrick loss-of-coolant accident analysis.

For B13.30 components (B-N-2), Enclosure 1 to Relief Request I5R-02, Revision 0, indicated that for the interior attachment welds that require VT-3 examination, the ASME Code, Section XI flaw evaluation criteria is employed (BWRVIP-48-A). This is also true for core spray piping bracket welds.

4.4 Additional Technical Findings

In addition to the above evaluations, the NRC staff also has the following findings:

- Although furnace-sensitized stainless steel vessel attachment welds tend to be more susceptible to intergranular stress-corrosion cracking, the NRC staff's approval of the BWRVIP-48-A I&E guidelines is an indication that the alternative monitoring of intergranular stress-corrosion cracking in this type of welds is acceptable.

- The licensee did not include BWRVIP-139, "BWR Vessel Internals Project, Steam Dryer Inspection and Flaw Evaluation Guidelines," to monitor active aging degradation in the steam dryer. In its RAI e-mail dated November 29, 2017, the NRC staff asked the licensee in RAI-3 to justify why BWRVIP-139 is not needed for either steam dryer hold-down brackets or steam dryer support brackets listed in Table 1 of its submittal under Item B13.30. In its RAI response dated December 11, 2017, the licensee explained that the guidance for the steam dryer hold-down or support brackets is contained in BWRVIP-48. Therefore, BWRVIP-139 is not needed for steam dryer support brackets listed in Table 1 under Item B13.30.

- The licensee did not include BWRVIP-42, Revision 1, "BWR Vessel Internals Project, Low Pressure Coolant Injection System (LPCI) Coupling Inspection and Flaw Evaluation Guidelines," to monitor active aging degradation in LPCI couplings. This is acceptable because BWRVIP-42, Revision 1, does not apply to older BWR/4 plants such as FitzPatrick.

- In addition to BWRVIP-25, some BWRVIP reports that are included in this relief request do not have approved "A" versions either. This is appropriate because use of the specific I&E guidelines in these BWRVIP reports for the ASME Code, Section XI, Examination Categories B-N-1 and B-N-2, Code Item Numbers B13.10 to B13.40 RVI components have already been accepted by the NRC staff in prior applications, as indicated in the June 30, 2014, safety evaluation for Grand Gulf Nuclear Station, Unit 1 (ADAMS Accession No. ML14148A262), and the March 10, 2016, safety evaluation for Clinton Power Station, Unit 1 (ADAMS Accession No. ML16012A344).

These findings clarified the scope of the applicable BWRVIP reports.

5.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request I5R-02, Revision 0, the proposed alternative 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) for Relief Request I5R-02, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year ISI interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

The NRC staff notes that if the licensee intends to take exceptions to, or deviations from, the NRC staff-approved BWRVIP inspection guidelines, this will require the licensee to revise and re-submit this relief request. The licensee shall obtain NRC staff approval for such exceptions prior to implementing the revised inspection guidelines for the FitzPatrick unit's RV interior surfaces, attachments, and core support structures.

All other requirements of the 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. Any ASME Code, Section XI RVI components that are not included in this relief request will continue to be inspected in accordance with the ASME Code, Section XI requirements

Principal Contributor: Simon Sheng

Date: May 30, 2018

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I5R-03, REVISION 0, TO USE ENCODED PHASED ARRAY ULTRASONIC EXAMINATION TECHNIQUES IN LIEU OF RADIOGRAPHY EXAMINATION

EXELON GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML17223A280 and ML17346A153, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Request I5R-03, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for its fifth 10-year inservice inspection (ISI) interval regarding the ferritic piping butt welds requiring radiography during repair/replacement activities at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick).

Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed to use an alternative that would allow the use of encoded phased array ultrasonic examination techniques (PAUT) in lieu of radiography (RT) examinations of ISI Class 1 and 2 ferritic piping repair/replacement welds required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for ISI of Nuclear Power Plant Components," at FitzPatrick. The regulation at 10 CFR 50.55a(z)(1) requires the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(g) of 10 CFR states, in part, that ISI of certain ASME Code Class 1, 2, and 3 systems and components be performed in accordance with Section XI of the ASME Code and applicable addenda incorporated by reference in the regulations, as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC 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 would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Section 50.55a of 10 CFR

allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The regulations require that inservice examination of components and system pressure tests conducted during the successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"), subject to the conditions listed in 50.55a(b). The applicable ASME Code of record for the fifth 10-year ISI interval for FitzPatrick is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

The guidance that the NRC staff considered in its review is NUREG/CR-7204, "Applying Ultrasonic Testing in Lieu of Radiography for Volumetric Examination of Carbon Steel Piping," September 2015 (ADAMS Accession No. ML15253A674), which provides an initial technical evaluation of the capabilities of phased-array ultrasonic testing to supplant traditional radiographic testing for detection and characterization of welding fabrication flaws in carbon steel welds.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 LICENSEE'S EVALUATION

3.1 Applicable Code Requirements

The regulation in 10 CFR 50.55a(b)(2)(xx)(B) states that "[t]he NDE [nondestructive examination] provision in IWA–4540(a)(2) of the 2002 Addenda of Section XI must be applied when performing system leakage tests after repair and replacement activities performed by welding or brazing on a pressure retaining boundary using the 2003 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section."

- Subarticle IWA-4540(a)(2) of the 2002 Addenda of the ASME Code, Section XI, states, in part, that "the nondestructive examination method and acceptance criteria of the 1992 Edition or later of Section III be met prior to return to service."
  Subarticle IWA-4540(a)(2) must be completed in order to perform a system leakage test in lieu of a system hydrostatic test. The examination requirements for ASME Section III circumferential butt welds are contained in the ASME Code, Section III, Subarticles NB-5200, NC-5200, and ND-5200. The acceptance standards for radiographic examination are specified in Subarticles NB-5300, NC-5300, and ND-5300.

- Subarticle IWA-4221 requires that items used for repair/replacement activities meet the applicable Owner's Requirements and Construction Code requirements when performing repair/replacement activities.

- Subarticle IWA-4520 requires that welded joints made for installation of items be examined in accordance with the Construction Code identified in the Repair/Replacement Plan.

3.2 Components for Which Relief is Requested

The ASME Code, Section XI, requires RT of ferritic piping butt welds during repair/replacement activities.

3.3 Licensee's Reason for Request

The licensee stated that implementation of the use of encoded PAUT in lieu of RT to perform the required examinations of the replaced welds would eliminate the safety risk associated with performing RT, which includes the planned exposure and the potential for accidental personnel exposure. The proposed alternative minimizes the impact on other outage activities normally involved with performing RT such as limited access to work locations and the need to control system fill status because RT would require a line to remain fluid empty in order to obtain adequate examination sensitivity and resolution. As part of its justification for the relief, the licensee also stated that encoded PAUT has been demonstrated to be adequate for detecting and sizing critical flaws and replacement of piping is periodically performed in support of the flow-accelerated corrosion (FAC) program as well as other repair and replacement activities.

3.4 Licensee's Proposed Alternative and Basis for Use

The proposed alternative includes a qualification program that the NRC staff determined is substantially similar to ASME Code Case N-831, "Ultrasonic Examination in Lieu of Radiography for Welds in Ferritic Pipe," approved by the ASME Section XI Standards Committee on October 20, 2016. The differences between the proposed alternative and ASME Code Case N-831 were limited to editorial changes that clarified the wording.

The encoded PAUT procedures, equipment, and personnel will be qualified using performance demonstration testing. The flaw acceptance standards for the PAUT examinations will consider all flaws to be planar and they are evaluated against the preservice acceptance standards of Subarticles IWB-3400, IWC-3400, and IWD-3400 of the ASME Code, Section XI, for ASME Code Class 1, 2, and 3 welds, respectively.

The licensee stated that the basis for the proposed alternative is that encoded PAUT is equivalent or superior to RT for detecting and sizing planar flaws. The examination procedure and personnel performing examinations are qualified by performance demonstration testing using representative piping conditions and flaws that demonstrate the ability to detect and size flaws that are both acceptable and unacceptable to the defined acceptance standards. The licensee also states that ultrasonic testing (UT) techniques are being used throughout the nuclear industry for examination of dissimilar metal welds and overlaid welds, as well as other applications, including piping replacements covered under ASME B31.1, "Power Piping, ASME Code for Pressure Piping, B31."

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information in Relief Request I5R-03, Revision 0. The NRC staff assessed the effectiveness of the use of UT in lieu of RT since 2009 through literature reviews, detailed evaluations of previous relief requests and proposed alternatives, and confirmatory experimental work to validate findings. Ultrasonic and radiography testing are volumetric inspection techniques that are commonly used to inspect welds in nuclear power plants and in other industries. Ultrasonic testing examinations differ from RT examinations as they use different physical mechanisms to detect and characterize discontinuities. These differences in physical mechanisms result in several key differences in sensitivity and discrimination capability. An assessment of the use of UT in lieu of RT is described in NUREG/CR-7204. This report included evaluation on the use of UT in lieu of RT for welded pipes and plates with thicknesses ranging from 0.844 inches to 2.2 inches.

Based on its review, the NRC staff finds that there is a sufficient technical basis for the use of UT in lieu of RT for ferritic steel welds. Given that UT can be effective, the NRC staff considered whether the proposed alternative applies UT in a way that provides reasonable assurance of finding structurally-significant flaws.

Important aspects of the licensee's proposed alternative include:

- The examination volume shall include 100 percent of the weld volume and the weld-to-base-metal interface.

- The electronic data files for the PAUT examinations will be stored as archival-quality records. In addition, hard copy prints of the data will also be included as part of the PAUT examination records to allow viewing without the use of hardware or software.

- Ultrasonic testing examination procedures shall be qualified by using either a blind or a non-blind performance demonstration using a minimum of 30 flaws covering a range of sizes, positions, orientations, and types of fabrication flaws. The demonstration set shall include specimens to represent the minimum and maximum diameter and thickness covered by the procedure.

- The flaw through-wall heights for the performance demonstration testing shall be based on the applicable acceptance standards for volumetric examination in accordance with Subarticles IWB-3400, IWC-3400, or IWD-3400 of the ASME Code, Section XI. At least 30 percent of the flaws shall be classified as acceptable planar flaws, with the smallest flaws being at least 50 percent of the maximum allowable size based on the applicable aspect ratio for the flaw.

- Ultrasonic testing examination personnel shall demonstrate their capability to detect and size flaws by performance demonstration using the qualified procedure. The demonstration specimen set shall contain at least 10 flaws covering a range of sizes, positions, orientations, and types of fabrication flaws.

- All flaws detected using angle-beam UT inspections will be treated as planar flaws and will be evaluated against the preservice acceptance standards in Subarticles IWB-3400, IWC-3400, and IWD-3400 of the ASME Code, Section XI, for ASME Code Class 1, 2, and 3 welds, respectively.

The NRC staff has authorized similar alternatives for other licensees, which include aspects similar to those listed above. The NRC staff finds that the use of performance demonstration for personnel and procedure qualification and the use of encoded data provide assurance that the PAUT methods will be sufficiently rigorous to detect and size flaws in the welds.

Currently, the licensee is required to use the RT acceptance standards in Section III of the ASME Code. Section III also provides UT acceptance standards; however, the licensee has requested to use the flaw acceptance standards in Section XI of the ASME Code as an alternative. The Section III RT and UT acceptance standards (Subarticles NB-5300, NC-5300, and ND-5300) require the inspector to detect and determine the type of flaw (e.g., porosity, lack of fusion, slag, incomplete penetration). While RT is effective at discerning between different flaw types, it is less capable than UT at detecting planar flaws such as cracks and lack-of-fusion defects. While Subarticles IWB-3400, IWC-3400, and IWD-3400 of Section XI of the ASME Code allow larger flaws than paragraphs NB-5330, NC-5330, and ND-5330 of Section III, the use of Section XI acceptance standards has proven effective for ISI of piping welds. The NRC staff finds that the use of the ASME Code, Section XI acceptance standards is appropriate for the proposed alternative, as the alternative is for repair/replacement activities, not new plant construction, and industry experience with Section XI acceptance standards has demonstrated their effectiveness.

Based on its review, the NRC staff finds that the use of the ASME Code, Section XI acceptance standards is appropriate for the proposed alternative, as the alternative is for repair/replacement activities, not new plant construction, and industry experience with Section XI acceptance standards has demonstrated their effectiveness. Therefore, the NRC staff concludes that the use of encoded PAUT qualified as proposed by the licensee for ferritic piping repair/replacement welds provides an acceptable level of quality and safety.

5.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request I5R-03, Revision 0, the proposed alternative 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) for Relief Request I5R-03, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year ISI interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

All other requirements of the 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. Any ASME Code, Section XI, RVI components that are not included in this relief request will continue to be inspected in accordance with the ASME Code, Section XI requirements.

Principal Contributor: Diane Render

Date: May 30, 2018

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I5R-04, REVISION 0, TO PERFORM INSERVICE ULTRASONIC

EXAMINATIONS OF REACTOR PRESSURE VESSEL THREADS IN FLANGE EXAMINATION

EXELON GENERATION COMPANY, LLC

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

1.0 INTRODUCTION

By letter dated August 10, 2017, as supplemented by letter dated December 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML17223A280 and ML17346A153, respectively), Exelon Generation Company, LLC (Exelon or the licensee) submitted Relief Request I5R-04, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for its fifth 10-year inservice inspection (ISI) interval regarding Examination Category B-G-1, Item Number B6.40 threads in flange locations at the James A. FitzPatrick Nuclear Power Plant (FitzPatrick).

Specifically, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee proposed to use an alternative to certain requirements of Section XI, "Rules for ISI of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) to perform in-service ultrasonic examinations of Examination Category B-G-1, Item Number B6.40, Threads in Flange for the fifth 10-year ISI interval at FitzPatrick. The regulations in 10 CFR 50.55a(z)(1) require the licensee to demonstrate that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Section 50.55a(g) of 10 CFR states, in part, that ISI of certain ASME Code Class 1, 2, and 3 systems and components be performed in accordance with Section XI of the ASME Code and applicable addenda incorporated by reference in the regulations, as a way to detect anomaly and degradation indications so that structural integrity of these components can be maintained.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a, or portions thereof, must be submitted and authorized by the NRC 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 would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety. Section 50.55a of 10 CFR

allows the NRC to authorize alternatives and to grant relief from ASME Code requirements upon making the necessary findings.

The regulations require that inservice examination of components and system pressure tests conducted during the successive 120-month inspection intervals (following the initial 120-month inspection interval) must comply with the requirements in the latest edition and addenda of the ASME Code, which was incorporated by reference in 10 CFR 50.55a(a) 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide (RG) 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"), subject to the conditions listed in 50.55a(b). The applicable ASME Code of record for the fifth 10-year ISI interval for FitzPatrick is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the Commission to authorize, the alternative requested by the licensee.

3.0 LICENSEE'S EVALUATION

3.1 Applicable Code Requirements

The ASME Code, Section XI, requires a volumetric examination technique with 100 percent of the flange threaded stud holes examined every ISI interval for reactor pressure vessel (RPV) threads in flange, Examination Category B-G-1, Item Number B6.40. The examination area is defined in Figure IWB-2500-12, "Closure Stud and Threads in Flange Stud Hole," of the ASME Code, Section XI.

3.2 Components for Which Relief is Requested

The ASME Code, Section XI, Class 1, Examination Category B-G-1, Item Number B6.40 threads in RPV flange locations at FitzPatrick.

3.3 Licensee's Reason for Request

The licensee stated that an evaluation of potential degradation mechanisms that could impact flange/threads reliability was performed. Potential types of degradation evaluated included pitting, intergranular attack, corrosion fatigue, stress-corrosion cracking, crevice corrosion, velocity phenomena, dealloying corrosion, general corrosion, stress relaxation, creep, mechanical wear and mechanical/thermal fatigue. Other than the potential for mechanical/thermal fatigue, there are no active degradation mechanisms identified for the threads in flange component. The licensee described maintenance activities it performs, each time the RPV closure head is removed to detect and mitigate general degradation prior to returning the reactor to service. Additionally, the licensee stated that the threads in the RPV flange are inspected for damage, cleaned, and lubricated prior to reinstallation of the RPV studs.

3.4 Licensee's Proposed Alternative and Basis for Use

The licensee is proposing to eliminate the examination of threads in the RPV flanges required by Examination Category B-G-1, Item No. B6.40 of the ASME Code, Section XI, for the duration of the fifth 10-year ISI interval, or until the NRC approves an applicable alternative in NRC Regulatory Guide 1.147 or other document. The licensee's request is based on an evaluation by the Electric Power Research Institute (EPRI) documented in EPRI Technical Report No. 3002007626 (EPRI report), "Nondestructive Evaluation: Reactor Pressure Vessel Threads in Flange Examination Requirements," March 2016 (ADAMS Accession No. ML16221A068). The licensee's submittals included information from the EPRI report regarding the generic stress analysis and the flaw tolerance evaluation, with additional plant-specific information to demonstrate applicability of the EPRI results to FitzPatrick. The submittals also included information from the EPRI report regarding operating experience and potential degradation mechanisms for threads in the RPV flanges.

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information in Relief Request I5R-04, Revision 0. The NRC staff focused its evaluation on the plant-specific applicability of the generic analyses contained in Section 6, "Stress Analysis and Flaw Tolerance Evaluation," of the EPRI report to FitzPatrick. Section 4, "Operating Experience," and Section 5, "Evaluation of Potential Degradation Mechanisms," of the EPRI report regarding operating experience and potential degradation mechanisms have already been accepted by the NRC staff as indicated in the safety evaluation dated June 26, 2017, enclosed in correspondence to Exelon for a relief request for 19 units (ADAMS Accession No. ML17170A013).

4.1 Stress Analysis

Stresses were determined from the finite element method analyses and used as input into the flaw tolerance analysis. The licensee described maintenance activities it performs each time the RPV closure head is removed to detect and mitigate general degradation prior to returning the reactor to service. The licensee stated that the threads in the RPV flange are inspected for damage, cleaned, and lubricated prior to reinstallation of the RPV studs. The NRC staff considers these activities beneficial to flaw detection and that they could potentially reduce flaw initiation. Therefore, the conservative nature of the stress and flaw tolerance analyses is verified periodically and maintained.

The NRC staff issued a safety evaluation dated January 26, 2017 (ADAMS Accession No. ML17006A109), on similar alternative requests that used the generic stress analysis for Vogtle Electric Generating Plant, Units 1 and 2, and Joseph M. Farley Nuclear Plant, Unit 1. Therefore, the current evaluation focuses on the licensee's demonstration of plant-specific applicability of this generic stress analysis to the RPV flange threads for FitzPatrick. In Relief Request I5R-04, the licensee summarized its plant-specific information in Table 1, "Comparison of JAFNPP Plant Parameters to Bounding Values Used in Analysis," and Table 2, "RPV Flange Thread Geometry."

4.1.1 *Evaluation of the Plant Parameters in Table 1 of Relief Request I5R-04*

In Table 1 of its submittal for Relief Request I5R-04, the licensee provided information on six key FitzPatrick plant parameters: number of studs, stud nominal diameter, RPV inside diameter at stud hole, flange thickness at stud hole, design pressure, and preload stress. This table shows that the stud nominal diameter for FitzPatrick is the same as that in the generic stress analysis, and the preload stress for FitzPatrick is less than the corresponding generic value, indicating that these two parameters are bounded by the generic analysis. As a result, only the other four parameters need to be evaluated. Three of them are used to calculate the operating pressure load per stud through the following equation:

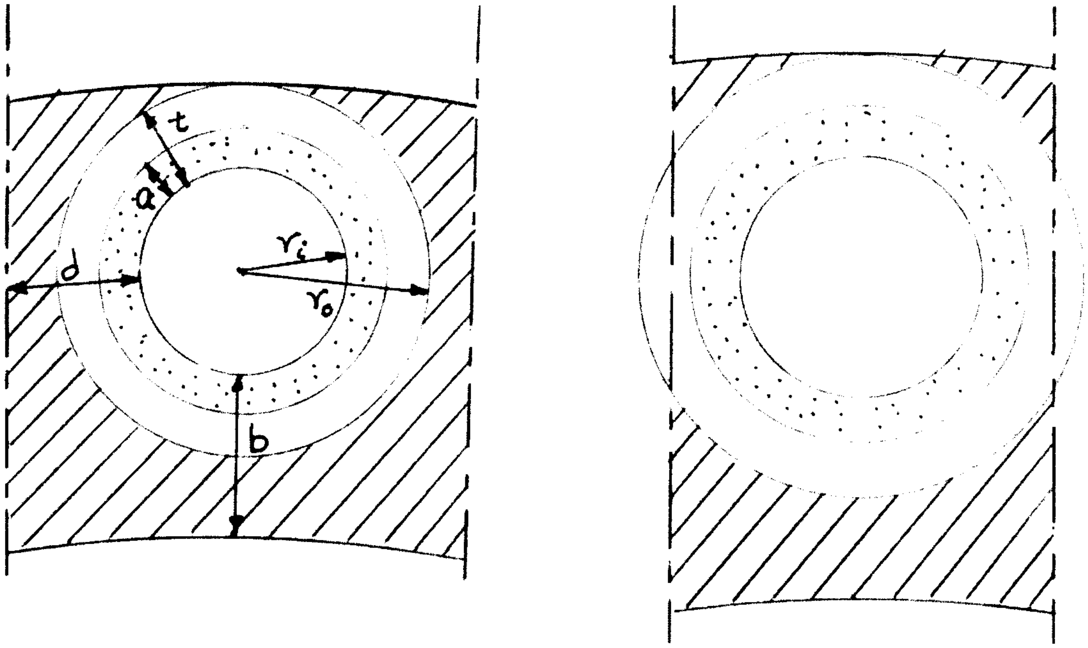
Load per stud = π(design pressure)(RPV inside diameter at stud hole)^2/(4xNo. of studs)

The NRC staff verified the licensee's calculation and confirmed that the load per stud for FitzPatrick is less than the corresponding generic value. Therefore, these three additional parameters are also bounded by the generic analysis. The last parameter (flange thickness at stud hole) leaves less RPV flange material in front of the critical crack front for FitzPatrick. Unfortunately, this is not evaluated by the licensee in the submittal.

4.1.2 *Evaluation of the Effect of a Seemingly Unbounded Parameter on Stresses*

In Table 1 of its submittal for Relief Request I5R-04, the RPV flange thickness at the stud hole for FitzPatrick is 13.5 inches versus 16 inches for the generic stress analysis and is not bounded by the generic analysis. This feature is common to all boiling-water reactors (BWRs). However, due to an oversight, the unboundedness was not evaluated in the June 26, 2017, safety evaluation for the relief request for the 19 units, which included many BWRs. The NRC staff has evaluated the significance of this seemingly unbounded parameter in this relief request. This safety evaluation should be referenced in future applications for BWRs. The FitzPatrick RPV has 52 studs around the circumference with an inner RPV radius of 109.4 inches versus 54 studs and 86.5 inches for the generic analysis. This makes FitzPatrick's circumferential thickness between stud holes approximately twice that of the generic analysis. Therefore, for the same preload, FitzPatrick's axial stresses in the material between the stud holes will be much less than the generic analysis and are, therefore, bounded by it. For comparison, the NRC staff provides schematics of the generic stress model and the FitzPatrick model in Figure 1 below with key dimensions (approximate values) listed in Table 1. The above observation is different in the RPV radial direction where the total flange radial thickness (considering both sides of the stud hole) is 7.5 inches for FitzPatrick versus 9 inches for the generic analysis. Since the preload contributes the most to the maximum K (85 percent to 96 percent according to Figure 6-1 of the EPRI report), examining the stress plot for the preload case in Figure 6-5 of the EPRI report is sufficient. Figure 6-5 shows that at the assumed crack location (i.e., ~ 15 percent down from the top edge of the figure), the axial stress is high in areas adjacent to the stud hole, but decreases rapidly to low and then negative values toward the RPV flange outer edge. This means that the stress pattern of the generic analysis is not sensitive to the reduced thickness between the stud hole and the outer edge of the FitzPatrick RPV flange. This observation applies to the reduced thickness between the stud hole and the inner edge of the FitzPatrick RPV flange, but of a much less concern because a significant portion of the thickness is under compressive stresses.

Figure 1. Schematics of the RPV Flange Stud Hole



FitzPatrick Generic

Table 1. Approximate Dimensions in Figure 1 for the FitzPatrick and Generic Models

	Relevant to Stress Model				Relevant to Fracture Model	
Key Dimensions	r_i (inches)	t (inches)	B (inches)	d (inches)	r_o (inches)	a (inches)
Generic	3.5	3.06	5.94	2.08	6.56	1.53
FitzPatrick	3	2.55	4.95	4.09	5.55	1.275

4.1.3 *Evaluation of the RPV Flange Thread Geometry in Table 2 of Relief Request I5R-04*

In Table 2 of its submittal for Relief Request I5R-04, the RPV flange thread geometry, which shows that for FitzPatrick, the pitch is eight threads per inch and the depth of threads is 0.06765 inch. In the generic stress analysis, the corresponding values are eight threads per inch and 0.06500 inch. The NRC staff evaluated differences of this magnitude in thread geometry on the final K results in the June 26, 2017, safety evaluation for the relief request for the 19 units and concluded that the impact is negligible. The same conclusion applies to FitzPatrick.

4.1.4 *Loads and Resulting Stresses*

The preload stress and the load per stud due to pressure for FitzPatrick are bounded by the generic analysis. The NRC staff found that the maximum heatup rate for FitzPatrick that is specified in Technical Specification 3.4.9, "RCS Pressure and Temperature (P/T) Limits," is also bounded by the generic heatup rate of 100 degrees Fahrenheit (°F)/hour. Please note that RCS P/T limits stands for reactor coolant system pressure-temperature limits. Therefore, all applied loads for FitzPatrick are bounded by the generic loads. Regarding the RPV flange stresses due to preload, the NRC staff's qualitative analysis indicated that flange axial stresses are not sensitive to the flange thickness at the stud hole. Based on its review, the NRC staff determined that the generic stress analysis results apply to FitzPatrick. However, since the driving force (i.e., the applied stress intensity factor, or the applied K) of the flaw tolerance analysis depends on the component geometry and the postulated flaw shape, the effect of the reduced RPV flange thickness on the flaw tolerance evaluation needs to be addressed. This is evaluated below in Section 4.2.

4.2. Flaw Tolerance Analysis

The licensee referenced the flaw tolerance analysis in the EPRI technical report as part of its basis to support the proposed alternative. The flaw tolerance analysis in the EPRI report, including the crack growth analysis, is based on the principles of linear elastic fracture mechanics. Similar to evaluation of the stress analysis, the NRC staff's current evaluation of the flaw tolerance analysis focuses on the effect to the generic analysis results due to the following FitzPatrick RPV flange information: (1) the flange material property and the bolt-up temperature and (2) the reduced flange thickness.

4.2.1 *Evaluation of the Effect of the RPV Flange Material Property and the Bolt-Up Temperature on Applied K*

In Table 4 of its submittal for Relief Request I5R-04, the licensee provided its RPV flange RT_{NDT} of 30 degrees Fahrenheit (°F) and bolt-up temperature of ≥ 60 °F for FitzPatrick. The NRC staff confirmed that "the information was obtained from plants as well as the NRC RVID2 database."

This information is consistent with that in the safety evaluation approving the relocation of FitzPatrick P/T limits from technical specifications to the licensee-controlled pressure temperature limits report in Amendment No. 292 for FitzPatrick, dated October 3, 2008 (ADAMS Accession No. ML082630365). Since preload is the dominant contributor to applied K, evaluation of the allowable K at the lowest P/T limits temperature is appropriate. Applying the ($T-RT_{NDT}$) of 30 °F to the fracture toughness (K_{IC}) equation in ASME Code, Section XI, Appendix A, the NRC staff verified the licensee's calculated K_{IC} value of 71 ksi$\sqrt{}$ in. Applying the acceptance criteria of ASME Code, Section XI, IWB-3600 (with safety margin of $\sqrt{10}$), the NRC staff verified that the allowed applied K would be 22.45 ksi$\sqrt{in}$, which is greater than all maximum K values in Table 3 for the preload case of the generic analysis. Based on its review, the NRC staff determined that FitzPatrick is bounded by the generic flaw tolerance analysis.

4.2.2 *Evaluation of the Effect of the Unbounded Parameter on Applied K*

The NRC staff examined the generic RPV flange model schematics (Figures 6-2 and 6-8 of the EPRI report) and the FitzPatrick geometry features discussed in Section 4.2 above and determined that a thick-cylinder model (with r_i and r_o shown in Figure 1) with an inner circumferential crack of a uniform depth ("a" shown in Figure 1) under uniform axial stresses can be used to estimate the adjustment factor for the maximum applied K at the crack tip close to the RPV flange outer edge for FitzPatrick. This adjustment factor is needed to account for the geometric differences between the FitzPatrick model and the generic model. This estimation is conservative because, as indicated in Figure 1, much more flange material close to the RPV flange outer edge outside the imaginary thick-cylinder is not considered in the FitzPatrick thick-cylinder model than in the generic thick-cylinder model. The applied K for the simplified thick-cylinder model and the key geometric parameters are summarized in the following table for both the generic and FitzPatrick cases.

Table 2. Key Parameters for the Applied K Calculation

Case	r_i/r_o	a/t	Applied K [1]	Geometry Adjustment Factor
Generic	0.534	0.5	2.605xstress	1.0
FitzPatrick	0.54	0.5	2.378xstress	0.91

A lower preload stress and a geometry adjustment factor of 0.91 for the applied K for FitzPatrick means that the applied K for FitzPatrick is bounded by the generic flaw evaluation, even though the thickness between the stud hole and the RPV flange outer edge for FitzPatrick is smaller than the generic model.

Regarding use of the crack growth analysis in the EPRI report to support the proposed alternative, the NRC staff considers it acceptable because the assumption of 400 occurrences of preload and 4,000 occurrences for heatup/cooldown for an 80-year period in the generic analysis are conservative for FitzPatrick.

4.0 CONCLUSION

As set forth above, the NRC staff determines that for Relief Request I5R-04, Revision 0, the proposed alternative 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) for Relief Request I5R-04, Revision 0. Therefore, the NRC staff authorizes the use of the alternative request for FitzPatrick for the fifth 10-year ISI interval, which began on August 1, 2017, and is currently scheduled to end on June 15, 2027.

All other requirements of the 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. Any ASME Code, Section XI, RVI components that are not included in this relief request will continue to be inspected in accordance with the ASME Code, Section XI requirements.

Principal Contributor: Simon Sheng

Date: May 30, 2018

[1] Page 27.7, "The Stress Analysis of Cracks Handbook," Second Edition, Hiroshi Tada, Paul Paris, and George Irwin.

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT – ISSUANCE OF RELIEF REQUESTS FOR ALTERNATIVES TO CERTAIN ASME CODE REQUIREMENTS (CAC NOS. MG0116, MG0117, AND MG0118; EPID L-2017-LLR-0083, EPID L-2017-LLR-0084, AND EPID L-2017-LLR-0085) DATED MAY 30, 2018

DISTRIBUTION:

Public
RidsNrrDorlLpl1
RidsACRS_MailCTR
RidsNrrPMFitzPatrick
RidsNrrLALRonewicz
RidsNrrDmlrMphb
RidsRgn1MailCenter
SSheng, NRR
DRender, NRR
MGray, R-I
JBowen, OEDO
RidsNrrLAJBurkhardt

ADAMS Accession No.: ML18039A854 *by safety evaluation dated

OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/LA	NRR/DMLR/MVIB/BC(A)*
NAME	BVenkataraman	LRonewicz (JBurkhardt for)	SRuffin
DATE	02/08/2018	05/21/2018	01/29/2018
OFFICE	NRR/DMLR/MPHB*	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/BC
NAME	DAlley	THood	JDanna
DATE	12/03/2017	05/16/2018	05/30/2018

OFFICIAL RECORD COPY