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

January 29, 2021

Mr. Frank R. Payne
Site Vice President
Energy Harbor Nuclear Corp.
Perry Nuclear Power Plant
P.O. Box 97, Mail Stop A-PY-A290
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 – ISSUANCE OF RELIEF RE: PROPOSED ALTERNATIVE REQUEST ASSOCIATED WITH FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (EPID L-2020-LLR-0004)

Dear Mr. Payne:

By letter dated January 6, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20006D984), as supplemented by letters dated June 2, 2020; July 15, 2020; and November 2, 2020 (ADAMS Accession Nos. ML20154K444, ML20197A212, and ML20308A436, respectively), FirstEnergy Nuclear Operating Company submitted relief requests (RRs) PT-001, Revision 3; IR-054, Revision 2; IR-056, Revision 3; and IR-060, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to the pressure test requirements in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," at Perry Nuclear Power Plant (Perry), Unit No. 1, associated with the fourth 10-year interval inservice inspection (ISI) interval.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative in RR IR-056, Revision 3, on the basis that the proposed alternative will provide an acceptable level of quality and safety.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative in RR IR-056, Revision 3, for the fourth 10-year ISI interval at Perry, Unit No. 1, which began on May 18, 2019, and is scheduled to expire on May 17, 2029.

All other ASME Code requirements for which relief was not specifically requested and approved in the subject request for relief remain applicable. The RRs identified as PT-001, Revision 3; IR-054, Revision 2; and IR-060, Revision 0, were approved separately by letter dated November 10, 2020 (ADAMS Accession No. ML20311A658).

If you have any questions, please contact the Perry project manager, at 301-415-2855 or by e-mail to Scott.Wall@nrc.gov.

Sincerely,

Nancy L. Salgado

Digitally signed by Nancy L. Salgado
Date: 2021.01.29 16:29:01 -05'00'

Nancy L. Salgado, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosure:
Safety Evaluation

cc: Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST IR-056, REVISION 3

PUMP AND VALVE INSERVICE TESTING PROGRAM

FOURTH 10-YEAR INTERVAL INSERVICE TESTING INTERVAL

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated January 6, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20006D984), as supplemented by letters dated June 2, 2020; July 15, 2020; and November 2, 2020 (ADAMS Accession Nos. ML20154K444, ML20197A212, and ML20308A436, respectively), FirstEnergy Nuclear Operating Company (FENOC) submitted Relief Requests (RRs) PT-001, Revision 3; IR-054, Revision 2; IR-056, Revision 3; and IR-060, Revision 0, to the U.S. Nuclear Regulatory Commission (NRC) for the use of alternatives to the pressure test requirements in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," at Perry Nuclear Power Plant, Unit No. 1 (Perry), associated with the fourth 10-year interval inservice inspection (ISI) interval, which began on May 18, 2019, and is scheduled to expire on May 17, 2029.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(1), the licensee requested to use the proposed alternative in RR IR-056, Revision 3, on the basis that the proposed alternative will provide an acceptable level of quality and safety.

The alternatives contained in PT-001, Revision 3; IR-054, Revision 2; and IR-060, Revision 0, were approved separately by letter dated November 10, 2020 (ADAMS Accession No. ML20311A658), and will not be discussed further in this safety evaluation (SE).

By order dated December 2, 2019 (ADAMS Accession No. ML19303C953), the NRC staff approved the direct and indirect license transfers for several FENOC-owned and operated plants, including Perry. By letter dated December 3, 2019 (ADAMS Accession No. ML19337B181), FENOC indicated that the entities taking control of the plants which had previously been referred to as New Hold Co, OwnerCo, and OpCo, would be named Energy Harbor Corp., Energy Harbor Nuclear Generation LLC, and Energy Harbor Nuclear Corp., respectively. Under this new set-up, Energy Harbor Corp. would indirectly own the plants as a parent company, Energy Harbor Nuclear Generation LLC would directly own the plants, and Energy Harbor Nuclear Corp. would have authority to operate the plants.

On February 20, 2020, FENOC informed the NRC (ADAMS Accession No. ML20054B733) that:

> Upon completion of the license transfer, Energy Harbor Nuclear Corp. will adopt and endorse the outstanding commitments, licensing actions, applications, and similar items on the aforementioned docket numbers. Energy Harbor Nuclear Corp. requests NRC continuation of the regulatory reviews and actions on these items.

On February 27, 2020, Energy Harbor Nuclear Corp. informed the NRC that the transaction closed on February 27, 2020, and that it adopted and endorsed the outstanding commitments, licensing actions, applications, and similar items on dockets submitted by FENOC on behalf of the licensees (ADAMS Accession No. ML20058D315). On February 27, 2020 (ADAMS Accession No. ML20030A440), the NRC staff issued Amendment No. 187 to reflect the license transfer. Accordingly, Energy Harbor Nuclear Corp. is now authorized to act as agent for Energy Harbor Nuclear Generation LLC and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility at Perry, Unit No. 1.

2.0 REGULATORY EVALUATION

The NRC regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g)(4), require that ASME Code Class 1, 2, and 3 components meet the ISI requirements, except the design and access provisions, set forth in Section XI of editions and addenda of the ASME Code to the extent practical, within the limitations of design, geometry, and materials of construction of the components. The interior accessible areas, welded attachments, and the welded core support structures in the reactor pressure vessel are categorized as ASME Code Class 1 components.

Section 50.55a(z) of 10 CFR states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used, when authorized by the NRC, if the licensee demonstrates: (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

The information provided by the licensee in support of the request for alternative to ASME Code requirements has been evaluated, and the bases for disposition are documented below.

Applicable Code Edition and Addenda

The applicable Code edition and addenda for the fourth ISI interval of Perry is the 2013 Edition of the ASME Code, Section XI.

3.1 Proposed Alternative IR-056, Revision 3

3.1.1 Applicable Code Requirement

Section XI of the ASME Code requires the visual examination (VT) of certain reactor vessel internal (RVI) components. These examinations are included in the ASME Code, Section XI,

Table IWB-2500-1, Categories B-N-1 (Interior of Reactor Vessel) and B-N-2 (Welded Core Support Structures and Interior Attachments to Reactor Vessels), and are identified with the following item numbers:

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

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

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

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

3.1.2 ASME Code Components Affected

ASME Code, Section XI, Class 1, Examination Category B-N-1 and B-N-2 item numbers:

- B13.10 – Vessel Interior
- B13.20 – Vessel Interior Attachments within Beltline Region
- B13.30 – Interior Attachments beyond Beltline Region
- B13.40 – Core Support Structure

3.1.3 Duration of the Alternative

The duration of the proposed alternatives is for the fourth 10-year ISI interval, which began on May 18, 2019, and is scheduled to end on May 17, 2029.

3.1.4 Reason for Request

The licensee stated that as an alternative to the ASME Code inspection requirements, use of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) inspection and evaluation (I&E) guidelines will avoid unnecessary inspections, while reducing radiological dose.

3.1.5 Licensee's Proposed Alternative

The licensee proposes to apply the BWRVIP I&E guidelines listed below to the affected ASME Code components identified in Table 1 of its submittal in lieu of the requirements of the ASME Code, Section XI, paragraph IWB-2500(a) and Table IWB-2500-1, including the examination method, examination volume, frequency, training, successive and additional examinations, flaw evaluations, and reporting.

- BWRVIP-03, “BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines”
- BWRVIP-18, Revision 2-A, “BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines”
- BWRVIP-25, “BWR Core Plate Inspection and Flaw Evaluation Guidelines”
- BWRVIP-26-A, “BWR Top Guide Inspection and Flaw Evaluation Guidelines”
- BWRVIP-27-A, “BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines”
- BWRVIP-38, “BWR Shroud Support Inspection and Flaw Evaluation Guidelines”
- BWRVIP-41, Revision 4-A, “BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines”
- BWRVIP-42, Revision 1-A,” BWR Vessel and Internals Project, Low Pressure Coolant Injection (LPCI) Coupling Inspection and Flaw Evaluation Guidelines”
- BWRVIP-47-A, “BWR Lower Plenum Inspection and Flaw Evaluation Guidelines”
- BWRVIP-48-A (applicable to all vessel attachment welds) and BWRVIP-48, Revision 1, “Vessel ID [Internal Diameter] Attachment Weld Inspection and Flaw Evaluation Guidelines” (applicable to vessel attachment welds and core spray piping brackets with revised inspection frequency)
- BWRVIP-76, Revision 1-A, “BWR Core Shroud Inspection and Flaw Evaluation Guidelines” (see note)
- BWRVIP-94NP, Revision 2, “BWR Vessel and Internals Project Program Implementation Guide”
- BWRVIP-100, Revision 1-A, “Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds”
- BWRVIP-138, Revision 1-A, “Updated Jet Pump Beam Inspection and Flaw Evaluation”
- BWRVIP-180, “BWR Vessel and Internals Project, Access Hole Cover Inspection and Flaw Evaluation Guidelines”
- BWRVIP-183-A, “Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines”

Licensee’s note: If flaw evaluations are required for BWRVIP-76 examinations, the fracture toughness values of BWRVIP-100, Revision 1-A, will be utilized.

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

3.1.6 Licensee’s Proposed Basis for Use

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

The BWRVIP recommended examinations specify locations that are known to be vulnerable to BWR relevant degradation mechanisms rather than “all surfaces.” The licensee stated that “the BWRVIP examination methods (including an enhanced visual examination VT-1 [EVT-1] or ultrasonic test [UT]) are superior to the ASME Code required VT-3 for flaw detection and characterization.” The licensee further stated that in most cases, the BWRVIP examination frequency is equivalent to or more frequent than the examination frequency required by the ASME Code. In cases where the BWRVIP examination frequency is less frequent than required by the ASME Code, the BWRVIP examinations are performed in a more comprehensive manner and focus on areas that are most vulnerable.

The licensee indicated that the enhanced flaw detection and characterization capability, with an equivalent or more frequent examination frequency, or with a less frequent examination frequency but with those examinations being performed in a more comprehensive manner, and using comparable flaw evaluation criteria, results in the BWRVIP criteria providing a level of quality and safety that is equivalent to or exceeds that provided by the ASME Code requirements. Thus, the licensee determined that use of these BWRVIP guidelines as alternatives to the ASME Code requirements provides an acceptable level of quality and safety.

3.1.7 Licensee’s Evaluation of the Inspection Results of the RVI Components

The licensee has evaluated the inspection results of the RVI components addressed in Electric Power Research Institute report, “Project No. 704-BWR Vessel and Internals Inspection Summaries for Fall 2017 Outages,” dated February 7, 2018 (ADAMS Accession No. ML18040A464). Based on its evaluation, the licensee has stated the following:

> Furnace-sensitized stainless-steel vessel attachment welds are inspected as required by the ASME Code and BWRVIP applicable guidelines. The sensitization status of the steam dryer and the feedwater support brackets has not been determined, and as such, they are assumed to be furnace-sensitized.

The licensee further stated that there are no Alloy 182 welds in Examination Category B-N-1 components. The licensee noted that Alloy 182 welds exist in the following Examination Category B-N-2 locations welds: (1) Shroud support (Weld H9), (2) Shroud support legs (Weld H12), and (3) Access hole cover.

The shroud support and shroud support leg inspections are specified in BWRVIP-38, and the access hole cover inspection is specified in BWRVIP-180. No additional augmented inspections are performed on the Alloy 182 welds outside of that defined in BWRVIP-38 and BWRVIP-180. There has been no cracking identified in the Alloy 182 welds at Perry. Furthermore, the licensee stated that it has implemented ISI and augmented inspection programs that are consistent with NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10” (ADAMS Accession No. ML19351E437). The licensee included, as part of its ISI program, six feedwater nozzle inner radii. The control rod drive return line at Perry has been cut and capped and, as such, is not subject to these examinations. The feedwater sparger flow holes and welds in the tees and arms are examined per NUREG-0619. Vessel attachment welds are examined by utilizing the requirements of BWRVIP-48-A.

3.1.8 Licensee's Evaluation of the Effects of Chemical Mitigation on the Inspection Criteria of the RVI Components

The licensee stated that it has implemented effective online noble chemical (OLNC) addition. The licensee provided information on the effectiveness of the OLNC in mitigating intergranular stress corrosion cracking (IGSCC). Effectiveness of the OLNC method is contingent on the licensee's compliance with the NRC staff's two conditions addressed in its letter, "Final Supplemental Safety Evaluation Related to BWRVIP-62-A, 'Boiling Water Reactor Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components With Hydrogen Injection,' Use of Online Noble Metal Chemistry in Boiling Water Reactors," dated July 6, 2018 (ADAMS Accession No. ML18142A019).

The licensee addressed the two conditions in the January 6, 2020, submittal: (1) electrochemical potential value of a stainless coupon measured with standard hydrogen electrode must be below the maximum limit addressed in the SE, (2) platinum deposit on the stainless coupon must meet the minimum amount specified in the staff's SE. The licensee complied with these conditions; therefore, the licensee has demonstrated that the implementation of OLNC addition is effective at Perry. The BWRVIP I&E guidelines addressed in the subject BWRVIP reports (Section 3.1.5 of this SE) allow inspection credit for the licensees that comply with the aforementioned staff's two conditions.

3.1.9 Licensee's Comparison of Inspection Criteria of the ASME Code, Section XI, Inspections with the Inspection Criteria Addressed in the Relevant BWRVIP Reports for the RVI Components

The licensee compared ASME Code, Section XI, Examination Category B-N-1 and B-N-2 requirements to BWRVIP guidance requirements. The licensee provided specific examples comparing the inspection requirements of the ASME Code, Section XI, Item Numbers B13.10, B13.20, B13.30, and B13.40 in Table IWB-2500-1, to the inspection requirements in the BWRVIP documents, including inspection methods. Furthermore, the licensee provided specific examples of how the I&E guidelines from various BWRVIP reports compare to the ASME Code requirements with respect to inspection scope, examination method, and frequency.

3.1.10 Deviations from the I&E Guidelines addressed in the BWRVIP-139-R1-A Report

With respect to taking deviation from the BWRVIP guidelines, the licensee stated that the only deviation from BWRVIP guidelines by Perry is related to BWRVIP-139-R1-A required examinations on the steam dryer, which is a non-ASME, non-safety-related component. Safe and event-free operation of the steam dryer during the extended 2-year period is supported by the structural integrity that was observed on the dryer during examinations completed in the 2009, 2011, 2013, 2015, and 2017 refueling outages.

3.1.11 Qualitative Risk Assessment for Extension of the Core Spray Piping Brackets Attachment Weld Examination Interval in Revision 1 to BWRVIP-48

In its supplemental letter dated November 2, 2020, the licensee submitted a technical evaluation developed by BWRVIP for implementing a revised inspection frequency for the primary and supplemental core spray piping brackets addressed in BWRVIP-48, Revision 1. The licensee's basis for implementing the proposed inspection frequency for the core spray piping brackets is addressed in the following sections. The BWRVIP stated that the vessel attachment welds for core spray piping brackets are susceptible to stress corrosion cracking (SCC), which is culminated by the combined effects of stresses in the weld, high oxygen concentration in the core spray systems due to lack of hydrogen water protection, and susceptible materials (i.e., stainless steel materials and Inconel Alloy 182 welds used in core spray piping bracket attachment welds). The degradation in these attachment welds due to SCC is confined to the groove weld joint between the weld buildup (for some locations) on the vessel and the core spray attachment, or welds between core spray piping brackets and vessel stainless steel clad material on the inside diameter of the vessel.

In the supplemental response, the licensee included the following items (addressed by BWRVIP) related to aging management of core spray piping brackets. The staff has addressed the details of the following items in Sections 3.2.5.1 through 3.2.5.7 of this SE. The items are:

(1) Historical Performance of Core Spray Piping Brackets
(2) SCC Susceptibility Discussion
(3) Weld Metal Susceptibility
(4) Heat Affected Zone (HAZ) on the (vessel side) Susceptibility
(5) State of Stress (weld and HAZ on the vessel side)
(6) SCC Risk Assessment
(7) Qualitative Risk Assessment Summary and Conclusion

3.2 NRC Staff Evaluation

The NRC staff found that the licensee has used referenced BWRVIP reports identified in its submittal as a technical basis for the proposed alternative because the BWRVIP I&E guidelines addressed in these reports would identify degradation in a timely manner and ensure that the integrity of the RVI components will be maintained. In addition, Perry's compliance with inspection criteria included in the reports would provide reasonable assurance that the age-related degradation in the RVI components will be identified in a timely manner. The first part of the staff evaluation (Sections 3.2.1 through 3.2.4) focuses on the aging degradation of RVI components. The second part of the staff evaluation (Section 3.2.5) focuses on the aging degradation of the core spray piping bracket welds.

3.2.1 Evaluation of the Inspection Results of the RVI Components

As part of its evaluation, the NRC staff reviewed the results of the reactor internals inspection history up until the most recent inspections in 2017 for Perry that were available. The staff noted that the RVI attachment welds at Perry were furnace sensitized, and therefore, stainless steel welds could be more susceptible to experience IGSCC. In addition, core shroud welds were made using Alloy 182 welding electrodes, and these welds are prone to experience IGSCC. Since no IGSCC cracking was observed at Perry, the staff determined that the licensee effectively has managed the active degradation due to IGSCC in components submerged under water using effective chemical mitigation, which is addressed in Section 3.2.2 of this SE.

Based on its review of these inspection summaries for Perry, the NRC staff finds that the licensee has adequately (1) identified the weld flaws (cracking); (2) taken 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) complied with scope expansion of inspections and subsequent inspections per the applicable BWRVIP reports. In addition, based on the review of the inspection results of the H9 and H12 Alloy 182 core shroud support welds, the staff noted that no cracking was observed in core shroud support welds.

The staff observed that BWRVIP-48 guidelines include more frequent inspections than ASME Code, Section XI criteria for the RVI components that are more susceptible to experience an active degradation (e.g., core spray piping welds, core shroud welds, etc.) These frequent inspections would enable the licensee to detect degradation in a timely manner. Therefore, subsequent inspections of the RVI components per the relevant BWRVIP I&E guidelines will provide reasonable assurance that any emerging degradation will be identified in a timely manner so that the licensee can take proper corrective actions to mitigate the degradation.

The NRC staff noted that many of the aforementioned BWRVIP reports addressed in Section 3.1.5 of this SE were approved by the staff. The BWRVIP-180 report was not submitted to the staff for review and approval. However, the NRC staff finds that the BWRVIP I&E guidelines addressed in the report meet or exceed the inspection frequency and techniques mandated by the ASME Code, Section XI ISI program. Hence, any active degradation can be identified in a timely manner, which ensures adequate structural integrity of the access hole cover will be maintained during the fourth ISI interval at Perry.

Furthermore, the staff noted that the licensee implemented ISI and augmented inspection programs for feedwater nozzles and control rod drive return line nozzle that are consistent with the NUREG-0619 program. Therefore, the staff concludes that implementation of inspections addressed in the NUREG-0619 program provides adequate assurance that any active degradation in these components will be detected in a timely manner, which allows the licensee to take corrective actions at Perry.

3.2.2 Effects of Chemical Mitigation on the Inspection Criteria of the RVI Components

Based on the information provided, the staff noted that OLNC methodology implemented at Perry complied with the staff's conditions addressed in the SE for BWRVIP-62-A. Therefore, the staff determined that the licensee adequately demonstrated that it implemented an effective OLNC method as part of chemical mitigation at Perry. The staff also noted that absence of any IGSCC occurrence in the most susceptible Alloy 182 welds confirms the effectiveness of chemical mitigation at Perry. Therefore, the staff concluded that the effective OLNC methodology would provide reasonable assurance that chemical mitigation is achieved in minimizing the IGSCC crack growth rates in RVI components in the water regions of the vessel at Perry.

3.2.3 Comparison of Inspection Criteria of the ASME Code, Section XI Inspections with the Inspection Criteria Addressed in the Relevant BWRVIP Reports for the RVI Components

The staff reviewed the licensee's evaluation of the differences in the inspection criteria between the ASME Code, Section XI, Examination Category B-N-1 and B-N-2 requirements with the relevant BWRVIP I&E guidelines for the various RVI components. The staff noted that the rigor of the inspection methods and inspection frequencies recommended in the BWRVIP I&E guidelines are more superior to the ASME Code, Section XI ISI criteria. For example, some of the most susceptible RVI components (i.e., core shroud welds) are inspected using UT, which is superior to a VT-3 examination as required by the ASME Code, Section XI components. Due to UT examinations, through-thickness defects can be discovered, and corrective action can be taken in timely manner. Additionally, better resolution of the indications using UT would be beneficial in assessing flaw evaluation. Therefore, the NRC staff concludes that the implementation of BWRVIP inspection guidelines in lieu of ASME Code, Section XI inspections would be more effective in monitoring the degradation in RVI components that are most susceptible to IGSCC.

3.2.4 Evaluation of the Issues Identified by the NRC Staff

During review of the submittal, the staff noted that the following issues require clarification: (1) reinspection criteria are not addressed for the lower plenum B-N-1 and B-N-2 RVI components in BWRVIP-47-A, and (2) the licensee's implementation of revised versions of BWRVIP reports (that were not reviewed by the staff) for substituting the ASME Code, Section XI inspection criteria for B-N-1 and B-N-2 RVI components.

In its supplemental letter dated July 15, 2020, the licensee stated that it had completed a one-time baseline inspection of the ASME Code, Section XI, B-N-1 and B-N-2 RVI components during the period 1999-2007. The licensee reiterated that it will perform inspections of the RVI components of the ASME Code, Section XI, B-N-1 and B-N-2 components when they become accessible for inspections during the fourth ISI interval. Based on the licensee's clarification, the staff concludes that the licensee will continue to perform ISI examinations of accessible areas (when available) of B-N-1 and B-N-2 RVI components of the lower plenum as required by the ASME Code, Section XI. The NRC staff finds this response acceptable and considers that this issue is closed.

With respect to the licensee's implementation of revised versions of BWRVIP reports that were not reviewed by the staff, the licensee stated the following:

> Energy Harbor Nuclear Corp. outlined in the January 6, 2020, submittal the specific BWRVIP reports that are being requested for approval as alternatives to the ASME Section XI Code requirements. Once the request is approved, Energy Harbor Nuclear Corp. initially implements and follows the inspection guidelines as written in the specified reports, including the revision number. In the event, that subsequent revisions are made by the BWRVIP to reports specified in the request, Energy Harbor Nuclear Corp. evaluates the nature of the revision and determines whether or not any changes were made to the requirements, guidelines, or scope of the report. If revisions were made that encourage or allow less conservative inspection frequencies, smaller inspection populations, and so forth, then subsequent relief is sought from the NRC to implement said guidelines if Energy Harbor Nuclear Corp. desires to implement them. If the revised guidelines are not used, the previous BWRVIP report outlined in the most recently approved request continues to govern.
>
> Under no circumstances would less conservative BWRVIP reports be implemented by Energy Harbor Nuclear Corp. at PNPP without explicit approval from the NRC.

The staff reviewed the aforementioned response and finds it acceptable because (a) the licensee will use the version of the BWRVIP report that is specified in the NRC-approved relief request, and (b) in accordance with 10 CFR 50.55a(z), a proposed alternative must be submitted and authorized prior to implementation.

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 the BWRVIP-48 guidelines, the licensee must submit a relief request to use a new version of the BWRVIP that is different from the version in the NRC-approved relief request. In addition, the licensee must continue to also meet the requirements of the version of the BWRVIP-48 guidelines that forms the basis for the NRC staff's authorized alternative to the requirements of 10 CFR 50.55a. The licensee may also choose to return to complying with the inspection requirements of the ASME Code of record for Perry. Thus, the NRC staff authorizes the licensee's proposed alternative based only on the specific revisions of the BWRVIP I&E guidelines proposed as an alternative in the licensee's submittal.

3.2.5 Qualitative Risk Assessment for Extension of the Core Spray Piping Brackets Attachment Weld Examination Interval in Revision 1 to BWRVIP-48

The staff reviewed the licensee's technical evaluation developed by BWRVIP for implementing a revised inspection frequency only for the primary and supplemental core spray piping brackets addressed in BWRVIP-48, Revision 1. BWRVIP stated that the vessel attachment welds for core spray piping brackets are susceptible to SCC, which is culminated by the combined effects of stresses in the weld, high oxygen concentration in the core spray systems due to lack of hydrogen water protection, and susceptible materials (i.e., stainless steel materials and Inconel Alloy 182 welds used in core spray piping bracket attachment welds). The aging degradation in these attachment welds due to SCC is confined to the groove weld joint between the weld buildup on the vessel and the core spray piping brackets, or welds between core spray piping brackets and vessel stainless steel clad material on the inside diameter of the vessel.

3.2.5.1 Historical Performance of Core Spray Piping Bracket Attachment Welds

BWRVIP stated that, to date, there have been 450 detailed visual examinations performed in the U.S. BWR fleet using EVT-1 for these welds. The staff reviewed the operating experience to date, which indicates that no SCC cracking was reported in the core spray piping bracket attachment welds to the vessel. This technique has the capability of identifying very small size cracking, and the inspection coverage obtained in these welds ranged from 80 percent to 100 percent for the primary core spray piping brackets. For supplemental brackets, the coverage ranged from 60 percent to 95 percent. Based on these observations, the staff determined that EVT-1 inspection techniques are effective in identifying small cracks, and in the absence of SCC in the core spray bracket attachment welds, the staff considers that if there were to be any SCC cracks in these welds, they would have been identified in the BWR fleet.

3.2.5.2 SCC Susceptibility Discussion

BWRVIP stated that main essential variables for causing SCC are: (1) environment, (2) material susceptibility, and (3) tensile stresses. The NRC staff reviewed the combined effects of these variables on the SCC, and the staff's evaluation is addressed below:

(1) Environment: The core spray piping bracket attachment welds are exposed to a highly oxidized environment with no effective protection from hydrogen. Therefore, a sample population more susceptible to SCC in core spray piping systems is frequently inspected during each outage in the BWR fleet.

(2) Material Susceptibility: Stainless steel base metals when they are sensitized due to heat generated in the welding process are more susceptible to SCC. This is called the sensitization process, and it occurs in the vicinity of HAZ in the base metal, adjacent to weld joint resulting in Chromium-depleted zone, which enhances the susceptibility to SCC. With respect to weld metal, stainless steel weld metal is less susceptible to SCC than Alloy 182 weld metal. The reason for this is due to the presence of ferrite in the stainless steel welds. Alloy 182 weld metal does not have ferrite metallurgical phase; therefore, it is more susceptible to SCC.

(3) Tensile Stresses: BWRVIP stated that only applied load to the core spray brackets is the dead weight of the core spray pipe. According to BWRVIP, the loading due to the dead weight is minimal; therefore, the predominant loading for causing SCC is the weld residual stresses of the core spray attachment brackets to the vessel. The staff considers that residual stresses associated with the weld joint are probably a predominant factor that could cause SCC. This is due to accumulation of residual stresses due to weld repairs that could potentially occur during the fabrication. This is an unknown factor that could accentuate the cracking during service.

3.2.5.3 Weld Metal Assessment

BWRVIP stated that core spray piping bracket attachment welds at Perry were fabricated using Type 308/308L austenitic stainless steel weld materials. As stated above, the stainless steel welds are less susceptible to SCC due to presence of ferrite metallurgical phase in the weld metal. The ASME Code, Section III, requires 5 percent of ferrite in the stainless steel weld metal. As a result, the operating experience, to date, indicated that cracks are not commonly observed in stainless steel welds unless they were subjected to heavy cold work during fabrication. Based on BWRVIP's recommendation, the licensees perform routine inspections in stainless steel weld joints and will focus on the inspections of HAZ areas where Chromium depletion occurs. For Perry, the core spray piping bracket attachment welds were fabricated with stainless steel weld metal, which has a higher amount of Chromium than the HAZ area in the stainless steel base metal. Therefore, the weld metal is less prone to SCC as opposed to the Chromium-depleted zone in HAZ areas. Since the piping bracket attachment welds are attached to the stainless steel weld buildup of the vessel or stainless steel vessel cladding, the licensee will inspect the HAZ areas of the piping bracket attachment welds in accordance with BWRVIP-48, Revision 1. The staff finds this type of inspection is acceptable because the licensees are inspecting the most susceptible areas of the weld joints.

3.2.5.4 HAZ (Vessel side) Material SCC Susceptibility

BWRVIP stated that for Perry, the core spray piping bracket attachment welds were fabricated with stainless steel weld metal, and the weld metal is less prone to SCC as opposed to the Chromium-depleted zone in the HAZ area. Since the piping bracket attachment welds are attached to the stainless steel weld buildup of the vessel or stainless steel vessel cladding, the HAZ areas of the piping bracket attachment welds will be inspected in accordance with BWRVIP-48, Revision 1. Therefore, the staff finds that BWRVIP's proposed inspections of the HAZ areas in the vessel side are acceptable because these areas are more prone to SCC than the weld metal.

3.2.5.5 State of Stress (Weld and HAZ on Vessel Side)

BWRVIP stated that the applied loads on the piping brackets during normal operations are low; therefore, it can be concluded that the predominant driving force for SCC can be the residual weld stresses associated with the double-welded groove joint of the core spray bracket welds. These bracket welds are addressed in Figure 2-8 of BWRVIP-48-A (ADAMS Accession No. ML043290381 (proprietary; non-publicly available)). BWRVIP stated that shrinkage stresses associated with the solidification cooling of the weld metal could enhance the residual stresses in the weld joint. This increase in the weld residual stress is directly proportional to weld joint restraint offered by thicker member in the weld joint assembly. Based on the weld design of the double-welded groove joint, as shown in Figure 2-8 in the BWRVIP-48-A report, the staff determined that the residual stress in this weld joint would be low. In addition, this type of weld joint configuration would facilitate the movement of the piping bracket and consequently the weld stresses will be low. Based on these factors, it can be concluded that the weld residual stresses in these weld joints are low enough to not cause any cracking due to SCC.

The staff noted that generally, the residual stresses can be minimized by using proper welding sequence in the top portion and bottom portion of the double groove to balance the stresses and minimize distortion. In addition, the piping bracket was not completely restrained, and it was completely free to move during the welding process. Therefore, in the absence of any weld joint restraint, it can be concluded that over the years, the SCC did not manifest in the core spray piping brackets attached to the vessel welds. Furthermore, the staff concludes that low to moderate weld heat input and limited weld repair could also have contributed to the resistance to SCC in the piping bracket vessel attachment welds. Based on the review, the staff determined that weld residual stresses in the attachment weld joints did not exceed the threshold limit to cause SCC. This conclusion can be based on the following potential reasons: (1) moderate weld heat input during the fabrication of the core spray bracket attachment welds, (2) limited weld restraint during welding, (3) proper weld sequencing, (4) minimum amount of grinding during welding, and (5) few weld repairs.

3.2.5.6 SCC Risk Assessment

Based on the review, the staff determined that even though the core spray piping systems are exposed to highly oxygenated reactor coolant water, no SCC to date was discovered in the core spray piping bracket welds to the vessel. This declining trend can be due to the replacement of the old piping with new piping which is more resistant to SCC. With respect to core spray piping brackets no cracks were observed because the weld joint configuration in these welds is different from the piping welds. The weld joint design used in the piping brackets provides less joint restraint compared to the piping weld. Therefore, it can be concluded that these welds are less likely to crack during the fourth ISI interval at Perry. The details of the effect of stress in these weld joints are discussed in Section 3.2.5.5 in this SE. The core spray piping brackets would be routinely inspected in accordance with the inspection criterion addressed in BWRVIP-48, Revision 1. The NRC staff determined that due to the time-dependent nature of SCC growth rates, if any new cracks were to occur in these welds, they would be identified, and corrective action would be taken by the licensee in a timely manner; therefore, the risk factor in not identifying the SCC in a timely manner is low.

3.2.5.7 Qualitative Risk Assessment and Conclusions

Even though there are no cracks identified thus far in the core spray piping bracket welds, the staff noted that BWRVIP implemented a conservative approach to monitor SCC in the core spray piping bracket welds at Perry. To date, a total of 450 inspections of core spray piping bracket welds was performed in the BWR fleet, and no cracks were found. In the absence of any SCC in these welds, the staff concludes that the licensee's proposed inspection frequency addressed in BWRVIP-48, Revision 1, for core spray piping bracket welds is acceptable based on the following observations:

(1) If any detection of SCC in core spray piping welds were to occur in any of the units in the BWR fleet, it would alert the industry and the staff alike to further evaluate the inspection frequencies in these welds. Field inspection data will be available to the fleet regularly because of different outages schedules of each BWR unit. If any SCC were to occur in the future, it provides opportunities for modifying the reexamination schedules of the core spray piping bracket welds.

(2) Previous inspection results addressed in BWRVIP-18, Revision 2-A, indicated that over the time interval, average crack growth rates due to SCC tend to decrease, which suggests that the cracking may be self-limiting. Since SCC is a time-dependent phenomenon based on item (1) above, any emerging cracks could be identified in a timely fashion in the BWR units.

(3) All the requirements addressed in BWRVIP-48-A, including reinspection and scope expansion criteria (Sections 3.2.2 and 3.2.3), apply for the core spray piping bracket welds. These inspection criteria provide reasonable assurance that the effective aging management program is implemented at Perry during the fourth ISI interval.

Based on these observations, the NRC staff concludes that the proposed inspection frequency addressed in the November 2, 2020, submittal provides reasonable assurance that the aging degradation due to SCC in the core spray piping bracket welds is adequately managed by the licensee during the fourth ISI interval. The NRC approval of the subject inspection frequency for core spray piping bracket welds addressed in BWRVIP-48, Revision 1, does not imply or infer the NRC's approval of the inspection criteria for generic use; and as such, it is approved on a plant-specific basis for Perry only for the duration of the fourth ISI interval. The NRC staff-approved inspection frequencies for core spray bracket vessel attachment welds addressed in BWRVIP-48, Revision 1, will supersede the previous inspection frequencies approved by the staff for these welds at Perry.

3.3 Technical Evaluation Summary

Based on the above considerations, the NRC staff finds that the implementation of the BWRVIP I&E guidelines specified in the licensee's proposed alternative will ensure the structural integrity of the RVI components with an acceptable level of quality and safety. Thus, the NRC staff authorizes only the BWRVIP I&E guidelines as an alternative to the ASME Code, Section XI ISI examination criteria. 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(z)(1).

4.0 CONCLUSION

As set forth above, the NRC staff finds that the proposed alternative described in RR IR-056, Revision 3, provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative in

RR IR-056, Revision 3, for the fourth 10-year ISI interval at Perry, which began on May 18, 2019, and is scheduled to end on May 17, 2029.

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

Principal Contributor: G. Cheruvenki

Date: January 29, 2021

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT NO. 1 – ISSUANCE OF RELIEF RE: PROPOSED ALTERNATIVE REQUEST ASSOCIATED WITH FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL (EPID L-2020-LLR-0004) DATED JANUARY 29, 2021

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