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

February 26, 2019

Ms. Tanya Hamilton
Site Vice President
Duke Energy Progress, LLC
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Rd.
M/C HNP01
New Hill, NC 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 RELIEF REQUEST I3R-18, REGARDING ALTERNATIVE REPAIR AND REPLACEMENT TESTING REQUIREMENTS FOR THE CONTAINMENT BUILDING EQUIPMENT HATCH SLEEVE WELD, INSERVICE INSPECTION PROGRAM FOR CONTAINMENT, THIRD TEN-YEAR INTERVAL (EPID L-2018-LLR-0081)

Dear Ms. Hamilton:

By letter dated June 4, 2018 (Agencywide Documents Access and Management System Accession No. ML18156A026), Duke Energy Progress, LLC (the licensee) submitted a request to the Nuclear Regulatory Commission (NRC) for the use of alternatives to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2007 Edition with 2008 Addenda requirements at Shearon Harris Nuclear Power Plant, Unit 1 (HNP).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the licensee requested to use the proposed alternative on the basis that the alternative provides an acceptable level of quality and safety. The licensee proposed to use a "bubble test-direct pressure technique" in accordance with the requirements of the ASME Code, Section V, Article 10, Appendix I, in lieu of the 10 CFR 50.55a requirement to perform a containment leak-rate test.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative was submitted in a timely manner and 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). Therefore, the NRC staff authorizes the use of proposed alternative I3R-18 for the one-time reactor vessel head replacement activity scheduled for the fall of 2019 refueling outage at HNP.

If you have any questions, please contact the Project Manager, Martha Barillas at 301-415-2760 or via e-mail at Martha.Barillas@nrc.gov.

Sincerely,

for

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosure: Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I4R-18 REGARDING ALTERNATIVE REPAIR AND REPLACEMENT

TESTING REQUIREMENTS FOR THE CONTAINMENT BUILDING EQUIPMENT HATCH

SLEEVE WELD, INSERVICE INSPECTION PROGRAM FOR CONTAINMENT,

THIRD TEN-YEAR INTERVAL (EPID L-2018-LLR-0081)

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

## 1.0 INTRODUCTION

By letter dated June 4, 2018 (Agencywide Documents Access and Management System Accession No. ML18156A026), Duke Energy Progress, LLC (the licensee) requested an alternative from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition with 2008 Addenda requirements at Shearon Harris Nuclear Power Plant, Unit 1 (HNP).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(z)(1), the licensee proposed an alternative to 10 CFR 50.55a(g)(4), as conditioned by 10 CFR 50.55a(b)(2)(ix)(J), for a Class MC pressure-retaining component. The licensee proposed to use a "bubble test – direct pressure technique" in accordance with the requirement of the American Society of Mechanical Engineers (ASME) Code, Section V, Article 10, Appendix I, in lieu of the 10 CFR 50.55a requirement to perform a containment leak-rate test. The licensee stated that this proposed alternative provides assurance of both containment structural integrity and leak-tight integrity prior to returning to service the Class MC pressure-retaining component after any major containment modification or repair/replacement activity. The licensee also stated that the proposed alterative to 10 CFR 50.55a provides an acceptable level of quality and safety.

## 2.0 REGULATORY EVALUATION

### 2.1 Component(s) for Which the Alternative is Requested

The steel containment building equipment hatch body ring and sleeve is a Code Class MC component, Seismic Category I, which functions as a pressure-retaining boundary. The

component has a 24-foot inside diameter, with a 1-1/4-inch-thick sleeve material that penetrates the cylindrical concrete containment building.

2.2 Proposed Alternative

In the alternative proposed in Relief Request (RR) I3R-18, the licensee proposed to perform a localized leakage test of the reinstalled weld area between the equipment hatch body ring and the containment sleeve using the "bubble test – direct pressure technique" in lieu of the Type A integrated leak rate test (ILRT) on the entire concrete containment structure as required by 10 CFR 50.55a(b)(2)(ix)(J). Specifically, the reinstalled weld area will be tested by pressurizing a test channel welded over the reinstalled weld area to 51.8 pounds per square inch gauge (psig) and using a film solution to detect leakage in accordance with ASME Code, Section V, Article 10, Appendix I. If leakage occurs, the weld will be repaired and the test will be repeated.

Prior to the localized leak rate test, the licensee will perform a 100-percent volumetric radiographic examination and a 100-percent magnetic particle test of the reinstalled weld area in accordance with ASME Code, Section III, Subsection NE. A channel strength test and simultaneous leakage test will be performed in accordance with the original design specification, CAR-SH-AS-1, paragraph 26.2, for the channel welded over the reinstalled weld area. For the solution film testing, the licensee proposed to use the "bubble test – direct pressure technique" in lieu of the halogen sniffer test specified in the original design specification.

The licensee states that the localized leakage bubble test on the pressure boundary weld area of the equipment hatch sleeve will provide a more effective examination than the Type A ILRT required by 10 CFR 50.55a(b)(2)(ix)(J). Therefore, the licensee requested an alternative to the 10 CFR 50.55a requirement, pursuant to 10 CFR 50.55a(z)(1), in that the proposed alternative provides an acceptable level of quality and safety.

2.3 Code Edition and Addenda of Record

The HNP equipment hatch was originally constructed as an ASME Code Class MC component in accordance with ASME Code, Section III, 1974 Edition with Winter 1975 Addenda.

The repair and replacement activities associated with temporary removal and reinstallation of the equipment hatch body ring under the HNP inservice inspection program, third interval of containment, will be performed in accordance with ASME Code, Section XI, 2007 Edition with 2008 Addenda.

2.4 Applicable Requirements

10 CFR 50.55a(g)(4) states in part,

*Components that are classified as Class MC pressure retaining components and their integral attachments, and components that are classified as Class CC pressure retaining components and their integral attachments, must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME BPV Code and addenda that are incorporated by reference in paragraph (a)(1)(ii) of this section, subject to the condition listed in paragraph (b)(2)(vi) of this section and the conditions listed in paragraphs (b)(2)(viii) and (ix) of this section, to the extent practical within the limitation of design, geometry, and materials of construction of the components.*

10 CFR 50.55a(b)(2)(ix)(J) states,

*(J)* Metal containment examinations: *Tenth provision. In general, a repair/replacement activity such as replacing a large containment penetration, cutting a large construction opening in the containment pressure boundary to replace steam generators, reactor vessel heads, pressurizers, or other major equipment; or other similar modification is considered a major containment modification. When applying IWE-5000 to Class MC pressure-retaining components, any major containment modification or repair/replacement must be followed by a Type A test to provide assurance of both containment structural integrity and leak-tight integrity prior to returning to service, in accordance with 10 CFR part 50, Appendix J, Option A or Option B on which the applicant's or licensee's Containment Leak-Rate Testing Program is based. When applying IWE-5000, if a Type A, B, or C Test is performed, the test pressure and acceptance standard for the test must be in accordance with 10 CFR part 50, Appendix J.*

10 CFR 50, Appendix J, paragraph IV.A states in part,

IV. Special Testing Requirements

> *(A) Containment modification. Any major modification, replacement* of a *component which is part* of *the primary reactor containment boundary, or resealing* a *seal-welded door, performed after the preoperational leakage rate test shall be followed by either* a *Type* A, *Type B, or Type* C *test as applicable for the area affected by the modification.*

ASME Code, Section XI, 2007 Edition with 2008 Addenda, IWE-5221(a) states,

*Except as noted in IWE-5224, a pneumatic leakage test shall be performed in accordance with IWE-5223 following repair/replacement activities performed by welding or brazing, prior to returning the component to service.*

ASME Code, Section XI, 2007 Edition with 2008 Addenda, IWE-5223 states, in part,

*IWE-5223.2 Boundaries. The test boundary may be limited to brazed joints and welds affected by the repair/replacement activity.*

*IWE-5223.4 Examination. During the pneumatic leakage test, the leak tightness of brazed joints and welds affected by the repair/replacement activity shall be verified by performing one of the following:*

> *(a) a bubble test-direct pressure technique in accordance with Section V, Article 10, Appendix I, or any other Section V, Article 10 leak test that can be performed in conjunction with the pneumatic leakage test*
>
> *(b) a Type A, B, or C Test, as applicable, in accordance with 10 CFR 50, Appendix J.*

2.5 <u>Reason for Request</u>

The licensee plans to replace the reactor pressure vessel head at HNP in the fall outage of 2019. The existing 15-foot bolted cover in the equipment hatch is not large enough to allow passage of the original reactor pressure vessel head or the replacement reactor pressure vessel head. The 24-foot diameter body ring welds to a 24-foot diameter penetration steel sleeve that

penetrates the containment. The licensee plans to cut the 24-foot diameter portion of the steel sleeve outside the containment wall to facilitate the movement of the old and new reactor vessel heads through the 24-foot opening created through the containment wall. Following completion of the reactor head replacement, the equipment hatch body ring will be re-welded to the penetration sleeve with a full penetration weld to restore the containment building equipment hatch to its original design configuration.

In RR I3R-18, the licensee proposed an alternative to the requirements in 10 CFR 50.55a(g)(4) with regard to the post-repair pressure testing requirements of the ASME B&PV Code, Section XI, 2007 Edition with 2008 Addenda, IWE-5000, as conditioned by 10 CFR 50.55a(b)(2)(ix)(J). Since the repair intends to restore the equipment hatch body ring weld to the containment penetration sleeve in accordance with ASME Code requirements, the licensee suggested that an effective post-repair test of the equipment hatch weld's leak-tight integrity can be performed by an alternate leakage test, which pressurizes only the area affected by the reinstallation weld.

### 2.6 Duration of the Alternative

The licensee requested the duration of the proposed alternative to be a one-time alternative for the ASME Code repair and replacement activity associated with the containment building equipment hatch, and shall be acceptable for the life of the plant, or until such time that a future repair and replacement activity of this nature is performed again, whichever comes first. The reactor vessel head replacement activity associated with this proposed alternative is scheduled for the fall outage of 2019.

### 2.7 Applicable Regulations

The licensee is proposing an alternative to 10 CFR 50.55a(g)(4), as conditioned by 10 CFR 50.55a(b)(2)(ix)(J), in accordance with 10 CFR 50.55a(z)(1). The U.S. Nuclear Regulatory Commission (NRC) staff may authorize alternatives to certain portions of 10 CFR as provided in 10 CFR 50.55a(z), which states, in part, that:

Alternatives to the requirements of paragraphs (b) through (h) of this section or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. The applicant or licensee must demonstrate that:

(1) *Acceptable level of quality and safety.* The proposed alternative would provide an acceptable level of quality and safety.

Given that the licensee has proposed an alternative to 10 CFR 50.55a(g)(4), as conditioned by 10 CFR 50.55a(b)(2)(ix)(J), and that 10 CFR 50.55a(z) permits the NRC staff to authorize alternatives to 10 CFR 50.55a paragraphs (b) through (h), the NRC staff finds that, subject to the following timeliness and technical evaluations, licensees may propose alternatives to 10 CFR 50.55a(g)(4) and that the NRC staff has the regulatory authority to authorize the licensee's proposed alternative in RR I3R-18.

### 2.8 Evaluation of Submittal Timeliness

Alternatives to 10 CFR 50.55a must be proposed within the timeframe specified in 10 CFR 50.55a(z), which states, in part, that:

A proposed alternative must be submitted and authorized prior to implementation.

The NRC staff evaluated the timeliness of this submission and has determined that the alternative was proposed in advance of the containment building equipment hatch being relied upon to meet its intended safety function and thereby meets the timeliness requirement of 10 CFR 50.55a(z). Therefore, the timeliness requirement of 10 CFR 50.55a(z) can be met by limiting the effective period of the NRC authorization to a period of time that begins at the time the alternative is authorized.

## 3.0 TECHNICAL EVALUATION

As previously stated in this safety evaluation, prior to authorizing the proposed alternative under 10 CFR 50.55a(z)(1), the NRC staff must find that the technical information provided in support of the proposed alternative is sufficient to demonstrate reasonable assurance that the structural integrity and intended safety function of the containment building equipment hatch are maintained for the life of the plant or until such time that a future repair and replacement activity of this nature is performed again, whichever comes first. If these criteria are met, the NRC staff finds that the proposed alternative to 10 CFR 50.55a(g)(4), as conditioned by 10 CFR 50.55a(b)(2)(ix)(J), will provide an acceptable level of quality and safety.

The NRC staff has reviewed the licensee's proposal and finds the following technical issues are important in determining if reasonable assurance exists that the structural integrity and intended safety function of the containment equipment hatch are maintained. Specifically, the technical issues evaluated are weld structural integrity and containment leak-tight integrity.

### 3.1 Weld Structural Integrity

The licensee is proposing to restore the equipment hatch original configuration by re-welding the equipment hatch body ring to the penetration sleeve using a full penetration weld, as shown in the RR Enclosure 1, Figure 4, "Reinstallation Weld", and by following the original design specifications and construction code requirements of ASME Code, Section III, Subsection NE. The HNP design specification CAR-SH-AS-1, "Containment Liner, Air Locks, and Hatch," and the Updated Final Safety Analysis Report Section 3.8.1.1.3.3, states that the equipment hatch was designed with sufficient material to allow up to six future removals and re-welding activities. The staff notes that, as described in the proposed alternative and as shown in Figure 3 of the RR Enclosure 1, the proposed re-welding activity represents the second removal and re-welding activity of the equipment hatch since its original installation. The first removal and re-welding activity was successfully performed by the licensee in 2001 during the steam generator replacement project. The staff finds the proposed re-welding activity acceptable because sufficient material exists in accordance with the HNP design specification to allow a second re-welding activity of the equipment hatch. The re-welding activity represents the second of the six re-welding activities for which this component was designed.

Once the equipment hatch is reinstalled, the licensee proposed to perform a post-weld examination on the equipment hatch repair which will include 100-percent radiographic examination of the weld and 100-percent magnetic particle testing of the weld to demonstrate 100-percent volumetric weld integrity. ASME Section III, Subsection NE requires all welds to be examined in accordance with the requirements of NE-5000 by qualified personnel. The staff finds the examination proposed by the licensee acceptable because the proposed examination is in accordance with the ASME Section III, Subsection NE, requirements and acceptance criteria to ensure weld integrity of the reinstalled component.

As described by the licensee, the repair and replacement activities associated with the temporary removal of the equipment hatch body and its reinstallation will be performed in accordance with the requirements of the 2007 Edition with 2008 Addenda of ASME Section XI, paragraph IWA-4411 that states that welding and installation activities shall be performed in accordance with the owner's requirements and in accordance with the construction code. The staff finds that the licensee has met this requirement because the equipment hatch will be restored to its original design configuration by performing all fabrication and installation activities using the original construction code of ASME Section III, Subsection NE, or as reconciled to a later edition, and by meeting the HNP original design specifications and licensing basis to ensure structural integrity of the reinstalled weld.

Based in the above, the staff has determined that the proposed re-welding activity and post-weld examinations in accordance with referenced code requirements and design specifications should ensure adequate structural weld integrity.

3.2 Leak-Tight Integrity

The licensee stated that once the equipment hatch body ring has been re-welded to the existing penetration sleeve, a leakage test in accordance with ASME Code, Section XI, 2007 Edition with 2008 Addenda, IWE-5223, as modified by 10 CFR 50.55a, Paragraph (b)(2)(ix)(J) would be required. Whereas IWE-5223.2 allows to limit the test boundary to the area impacted by the repair/replacement activity, 10 CFR 50.55a(b)(2)(ix)(J) requires an ILRT (also referenced as a Type A test in 10 CFR 50, Appendix J) as the modification meets the definition of a major modification provided therein (i.e.,"... cutting a large construction opening in the containment pressure boundary to replace steam generators, reactor vessel heads, ..."). The licensee stated that an effective post-repair test of the equipment hatch weld's leak tight integrity can be performed by an alternate leakage test, which pressurizes only the area affected by the reinstallation weld.

HNP has adopted 10 CFR Part 50, Appendix J, Option B Performance Based Requirements for ILRT (Type A) of the containment. The local leak tests (Type B and Type C) remained under the purview of 10 CFR Part 50, Appendix J, Option A "Prescriptive Requirements."

The last Type A test at HNP was conducted in 2012. Based on the performance-based requirements, the next Type A would not be due at least until 10 years later, except under circumstances of previous failures of Type A test or major activities such as cutting a large construction opening, such as the cutting of the equipment hatch opening that is necessary for moving the reactor pressure vessel heads in and out of the containment during the 2019 refueling outage. The licensee's current request applies to relief from having to perform a Type A test during the 2019 refueling outage.

The Type B tests for the equipment hatch O-ring seals at HNP are tested every refueling outage to meet the 24-month frequency requirement. The licensee indicated that this test will be conducted per the requirement of HNP Surveillance Requirement 4.6.1.2.a.

In lieu of the post-repair 10 CFR Part 50, Appendix J, Type A test, the licensee is proposing a localized leakage test on the equipment body ring to the containment sleeve reinstallation weld area. A leak chase channel will be welded over the equipment channel reinstallation weld with a screwed half coupling to allow pressurization of the reinstallation weld area. The relief request provided the basis, features, and slight variations of the test as follows:

ASME Code, Section XI, 2007 Edition with 2008 Addenda, IWE-5221(a) states that a pneumatic test shall be performed in accordance with IWE-5223 following repair/replacement activities performed by welding and brazing prior to returning the component to service. ASME Code, Section XI, 2007 Edition with 2008 Addenda, IWE-5223 states:

IWE-5223.1 Pressure. The pneumatic leakage test shall be conducted at a pressure between 0.96 Pa and 1.10 Pa, except when otherwise limited by plant technical specifications, where Pa is design basis accident pressure.

According to Technical Specification 6.8.4.k, Pa for HNP is 41.8 psig, thus 1.10 Pa is approximately 45.6 psig. However, the licensee prefers to perform the test at a higher pressure of 51.8 psig due to a desire to conform to the original design specification CAR-SH-AS-1. Excerpts from the CAR-AH-AS-1 were provided in Enclosure 2 to the RR. The basis for the higher pressure of 51.8 psig is 1.15 x Pd, where Pd is the design pressure of 45 psig. The staff finds this variance from IWE-5223.1 acceptable because it is a local test for the welds in the restored area of the hatch, and the hatch was previously subjected to the higher pressure by the equipment supplier.

IWE-5223.2 Boundaries. The test boundary may be limited to brazed joints and welds affected by the repair/replacement activity. The proposed test conforms to this requirement.

IWWE-5223.4 Examination. During the pneumatic leakage test, the leak-tightness of the brazed joints and welds affected by the repair/replacement activity shall be verified by performing a bubble test described as a direct pressure technique in accordance with Section V, Article 10, Appendix I, or other Section V, Article 10 leak test that can be performed in conjunction with the pneumatic test. An alternative to the bubble test can be a Type A, B, or C test, as applicable, in accordance with 10 CFR Part 50, Appendix J. The licensee is proposing to perform a bubble test. The bubble test will also satisfy the equipment supplier requirement to perform solution film testing. The pressure applied during the test will be 51.8 psig. If leakage occurs, the weld will be repaired and retested. Thus, the acceptance criteria for the local leakage will essentially ensure zero leakage at the weld area.

The staff also considered the licensee's previous experience with the proposed test as part of the testing performed during the 2001 steam generator replacement project, where the equipment hatch was removed and reinstalled. However, a post-modification ILRT was not a requirement at that time. The licensee stated that the test configuration used in 2001 is similar to the test configuration proposed in the relief request. The licensee stated that a Type A ILRT was performed in 2012 with no issues identified with leakage at the equipment hatch circumferential weld. The test results were found acceptable with approximately 42-percent margin remaining.

The staff has determined that, from a containment leakage testing perspective, the proposed test is in accordance with referenced code requirements and is functionally equivalent to the Type A test. The local leak-rate test proposed by the relief request should ensure essentially no leakage through the tested reattachment weld area. The reattachment weld will be subsequently tested as part of the periodic Type A tests.

3.3 Assessment of Quality and Safety

Based on the analysis discussed in sections 3.1 and 3.2 of this SE, the NRC staff finds that reasonable assurance exists that the structural integrity and intended safety function of the equipment hatch are maintained. Therefore, the licensee's proposed alternative in RR I3R-18 provides an acceptable level of quality and safety.

## 4.0 CONCLUSION

As discussed above, the NRC staff has determined that the proposed alternative was submitted in a timely manner and provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) and is in compliance with the ASME Code requirements. Therefore, the NRC staff authorizes the use of proposed alternative in RR I3R-18 for the one-time reactor vessel head replacement activity scheduled for the fall of 2019 refueling outage at HNP.

All other 10 CFR 50.55a requirements for which the relief request was not specifically requested and authorized in this proposed alternative remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors:
J. Lopez,
N. Karipineni

Date: February 26, 2019

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 RELIEF REQUEST I3R-18, REGARDING ALTERNATIVE REPAIR AND REPLACEMENT TESTING REQUIREMENTS FOR THE CONTAINMENT BUILDING EQUIPMENT HATCH SLEEVE WELD, INSERVICE INSPECTION PROGRAM FOR CONTAINMENT, THIRD TEN-YEAR INTERVAL (EPID L-2018-LLR-0081) DATED FEBRUARY 26, 2019

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