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

June 25, 2018

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2 – RELIEF REQUEST (RR)-12 REGARDING PROPOSED ALTERNATIVE TO ASME CODE CASE N-729-4 EXAMINATION FREQUENCY REQUIREMENTS (EPID L-2017-LLR-0089)

Dear Mr. Kapopoulos:

By letter dated September 22, 2017, Duke Energy Progress, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of an alternative to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI requirements at the H. B. Robinson Steam Electric Plant Unit No. 2. In RR-12, the licensee proposed an alternative to the examination frequency requirements of Code Case N-729-4, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure Retaining Partial-Penetration Welds, Section XI, Division 1."

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 on the basis that the alternative would 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 of the regulatory requirements set forth in 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes RR-12 at the H. B. Robinson Steam Electric Plant Unit No. 2 until the 32nd refueling outage, which is scheduled to commence in September of 2020.

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

If you have any questions, please contact the Project Manager, Dennis Galvin, at 301-415-6256 or Dennis.Galvin@nrc.gov.

Sincerely,

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-261

Enclosure:
Safety Evaluation

cc: Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST (RR)-12 REGARDING PROPOSED ALTERNATIVE TO

ASME CODE CASE N-729-4 EXAMINATION FREQUENCY REQUIREMENTS

DUKE ENERGY PROGRESS, LLC

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated September 22, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17269A016), Duke Energy Progress, LLC (the licensee) submitted a request to the U.S. Nuclear Regulatory Commission (NRC) for the use of an alternative to certain American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," requirements at the H. B. Robinson Steam Electric Plant Unit No. 2 (Robinson). In Relief Request (RR)-12, the licensee proposed an alternative to the examination frequency requirements of Code Case N-729-4, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads with Nozzles Having Pressure Retaining Partial-Penetration Welds, Section XI, Division 1."

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-12, in lieu of the volumetric or surface examination frequency for the reactor pressure vessel head nozzles of Table 1 in ASME Code Case N-729-4, on the basis that the alternative examination frequency provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components is to be performed in accordance with ASME Code, Section XI, and applicable editions and addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the NRC.

Pursuant to 10 CFR 50.55a(g)(6)(ii), the NRC may require the licensee to follow an augmented ISI program for systems and components for which the NRC deems that added assurance of structural reliability is necessary. The regulations in 10 CFR 50.55a(g)(6)(ii)(D) require, in part, that "[a]ll licensees of pressurized water reactors must augment their inservice inspection program with ASME Code Case N-729-4, subject to conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4)…."

The regulations in 10 CFR 50.55a(z) state that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates that (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty, without a compensating increase in the level of quality and safety.

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 NRC to authorize, the proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Components Affected

The affected components are ASME Class 1 PWR reactor vessel closure head (RVCH) nozzles and partial-penetration welds fabricated with Alloy 690/52/152 materials. Each of these nozzles and associated welds are categorized as Item B4.40 in ASME Code Case N-729-4, Table 1, to identify the volumetric inspection frequency requirement for these components.

3.2 Inservice Inspection Interval and Applicable Code Edition and Addenda

Robinson is currently in its fifth 10-year ISI interval, which began July 22, 2012, and ends July 30, 2021. The ASME Code of Record for the fifth 10-year ISI interval is the 2007 Edition with 2008 Addenda.

3.3 Code Requirement for Which Relief is Requested

The regulation in 10 CFR 50.55a(g)(6)(ii)(D) requires that licensees augment their ISI program in accordance with ASME Code Case N-729-4 subject to the conditions specified in paragraphs (2) through (4) of 10 CFR 50.55a(g)(6)(ii)(D). ASME Code Case N-729-4, Table 1, Inspection Item B4.40, requires volumetric or surface examinations be performed within one inspection interval (nominally 10 calendar years) for a replaced RVCH with primary water stress corrosion cracking (PWSCC) resistant nozzles and weld materials.

3.4 Proposed Alternative

The licensee requests to extend the frequency of the volumetric or surface examination of the Robinson RVCH of Table 1, Item B4.40 of ASME Code Case N-729-4, for one additional operating cycle beyond that approved by the NRC in accordance with Robinson RR-11 (ADAMS Accession No. ML15021A354). This request would extend the volumetric or surface examination to the 32nd refueling outage, which is scheduled to commence in September of 2020. The licensee requests the total time deferral of approximately 5 years (an additional 2 years beyond the previously approved deferral of 3 years (ADAMS Accession No. ML15021A354)) for Robinson for the purpose of aligning with the scheduled Robinson refueling outages.

3.5 Licensee's Basis for Proposed Alternative

The licensee stated that the Electric Power Research Institute (EPRI) published final report "Materials Reliability Program: Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375)," dated

February 2014 (ADAMS Accession No. ML14283A046), provides a technical justification to extend the volumetric/surface examination interval of the RVCH nozzle penetrations from 10 years to 20 years. The licensee also stated that the ASME Code Case Committee adopted the revised volumetric/surface examination of two inspection intervals (20 years) in ASME Code Case N-729-5. In summary, the licensee proposed to extend the inspection interval from the nominal 10 calendar years to 15 calendar years, based on plant service experience and factor of improvement (FOI) studies using laboratory data.

The licensee further used the parameters of crack growth rate defined by ASME Code Case N-729-4 and calculated the FOI needed to support the proposed alternative. The licensee found that an FOI of 6.6 was necessary for its replacement RVCH with Alloy 690/52/152 materials to support extending the inspection interval to 15 calendar years. The licensee noted that the PWSCC crack growth rates for Alloy 690/52/152 materials are significantly lower than those of Alloy 600/82/182 materials, and therefore, merit a much longer inspection interval than required by ASME Code Case N-729-4. Therefore, in order to show that the inspection interval extension provides reasonable assurance of structural integrity, the licensee showed that a crack growth rate FOI of 6.6 was acceptable by comparing the available crack growth rate curves of Alloy 600 materials to the available crack growth rate data for Alloy 690 materials. The licensee provided this comparison with additional analysis and calculations by Dominion Engineering, Inc., as noted in the attachment to the submittal entitled, "Technical Note, Assessment of Laboratory PWSCC Crack Growth Rate Data Compiled for Alloys 690, 52 and 152 with Regard to Factors of Improvement (FOI) versus Alloys 600 and 182," TN-5696-00-02, Revision 0, March 2015.

The TN-5696-00-02 report utilizes laboratory crack growth rate test data for Alloy 690/52/152 presented in MRP-375 and data from the summary report prepared by two NRC contractors, Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL), dated October 30, 2014 (ADAMS Accession No. ML14322A587). TN-5696-00-02 concludes that the resistance to PWSCC by the materials used for the nozzles and associated welds in the replacement RVCH at Robinson support an FOI of greater than 12 based on crack growth rate alone. The licensee's conclusion is that this FOI of 12 is greater than the FOI of 6.6 necessary for an extension of the volumetric inspection frequency to 15 calendar years. Therefore, the licensee explained that the available laboratory data supports the FOI implied by the requested extension period. The licensee concluded that the proposed alternative revised volumetric/surface examination interval provides an acceptable level of quality and safety as conditioned by 10 CFR 50.55a(z)(1).

3.6 <u>NRC Staff Evaluation</u>

In evaluating the technical sufficiency of the licensee's proposed alternative to defer the Robinson RVCH nozzle penetration and associated J-groove weld volumetric/surface examination interval to 15 calendar years, the NRC staff considered the licensee's basis for use of the proposed alternative in accordance with 10 CFR 50.55a(z)(1), on the basis that the alternative examination frequency provides an acceptable level of quality and safety.

The inspection frequencies developed in Code Case N-729-4 for RVCH penetration nozzles made of Alloy 690/52/152 were developed based, in part, on a conservative assessments of the limited crack growth rate data and operating experience of these materials. The licensee's primary technical basis is that the available crack growth rate data is now sufficient to justify a longer inspection interval, and demonstrate a sufficient FOI of these materials as compared to the Alloy 600/82/182 materials. This FOI would then provide the basis for the extension of the

ISI frequency requested by the licensee in its proposed alternative. The licensee calculated that it needed a FOI of 6.6. The NRC staff independently verified that the licensee's requested alternate inspection interval of 15 calendar years is reasonably bounded by the licensee's calculated FOI of 6.6, by using the parameters defined by ASME Code Case N-729-4 and using Robinson's upper head estimated operating temperature of 599.75 degrees Fahrenheit.

In evaluating the licensee's technical basis for the proposed alternative, the NRC staff notes that the licensee uses MRP-375. MRP-375, in part, summarizes numerous Alloy 690/52/152 crack growth rate data from various sources to develop FOIs for the crack growth rate equations provided in MRP-55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials," and MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds." While the NRC staff finds the licensee's assertions and/or interpretations to be reasonable, MRP-375 is not an NRC-approved document. Additionally, the NRC staff has not validated all of the data reported in MRP-375. Therefore, the NRC staff does not consider it appropriate to use all of the data from MRP-375 in its review of the licensee's relief request. A more detailed review of the data provided in MRP-375 has been performed by an international group of experts as part of an Alloy 690 Expert Panel. This report entitled, "Materials Reliability Program: Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP-386)," is still under review at the NRC. It is expected that the full review will be complete in the fall of 2018.

In the interim, the NRC staff's review of the proposed alternative will rely upon Alloy 690/52/152 crack growth rate data from two NRC contractors, PNNL and ANL. The data is documented in the PNNL and ANL summary report, dated October 30, 2014. The majority of the data from PNNL and ANL for Alloy 690 test samples was generally consistent with the overall data presented in MRP-375, and also supports the licensee's use of an FOI value of 6.6. The only PNNL and ANL data that was not bound by the licensee's FOI was that associated with weld dilution specimens. This means that certain crack growth rate tests of weld dilution samples would have an FOI of less than 6.6 versus the crack growth rate curves for Alloy 600 weld materials. Therefore this data would not support the licensee's requested inspection frequency extension. The NRC staff considered the impact of this limitation.

The NRC staff considered the PNNL and ANL weld dilution data and its significance to address the licensee's requested inspection frequency extension. It should be noted that the PNNL and ANL data summary report includes very limited weld dilution testing, and no general conclusions have been reached regarding the use or applicability of the weld dilution data. This was because there was a high variability in the data, including results of low or no growth in some cases, and in the cases of fast growth, only over limited distances. However, the data was included in the report, as the report provides a summary of all data collected to be assessed by the NRC staff.

Ultimately, the NRC staff chose to exclude the weld dilution zone data from this analysis due to (1) the limited number of data points available, (2) the variability in the results, and (3) the limited area of continuous weld dilution for potential flaws to grow through. For example, in the case of dilution zone measured crack growth rates, a flaw would have to travel at the interface between the J-groove weld and the low alloy steel interface in the RVCH. Since such a hypothetical flaw would have to grow through a region of stainless steel, non-diluted Alloy 52/152, or a dilution of Alloy 52/152 and stainless steel (a condition for which dilution testing has shown limited or no growth), the NRC staff finds it is not reasonable for a significantly increased probability of leakage or component failure. Therefore, the NRC staff considers the impact of these weld dilution zone crack growth rates to be of limited relevance for this specific relief request.

The NRC staff does note that dilution zone crack growth rate testing is ongoing. Exclusion of the crack growth rate data in the weld dilution zone may be reevaluated as additional data becomes available, a better understanding of the existing data is obtained, or if a longer extension of the inspection interval is requested in the future. However, even the latest EPRI report, MRP-368, "Materials Reliability Program: Risk Assessment of ASME Section XI Appendix G Pressure-Temperature Limit Curve Methodologies," states, "A lack of sufficient data for each weld interface type in the current crack growth rate database meant that the effects of weld interfaces and chromium dilution on PWSCC growth rates could not be quantitatively assessed as part of this effort." Therefore, no near term conclusion is expected to resolve this issue beyond the current NRC staff assessment above.

Given the above discussion of the outlying weld dilution data, the NRC staff finds that the licensee's needed FOI of 6.6 is allowable. Therefore, the NRC staff finds that the licensee's proposed alternative is justified and bounded by the relevant available data included in the PNNL and ANL report, thus providing an acceptable level of quality and safety.

The NRC staff finds that the licensee's analyses provided sufficient technical justification to support the proposed alternative of extending the volumetric/surface inspection interval for Robinson's replacement RVCH to 15 calendar years. The NRC staff finds that the proposed alternative does not pose a higher risk than the inspection frequency associated with an RVCH with Alloy 600/82/182 nozzles and associated J-groove welds that are inspected at intervals as specified in 10 CFR 50.55a(g)(6)(ii)(D). Hence, the NRC staff finds the licensee's technical basis to be acceptable. Therefore, based on the above evaluation, the NRC staff finds that the proposed alternative provides an acceptable level of quality and safety as required by 10 CFR 50.55a(z)(1).

4.0 CONCLUSION

As set forth above, the NRC staff has determined that the alternative method proposed by the licensee in RR-12 provides an acceptable level of quality and safety for the examination frequency requirements of the RVCH at Robinson. 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 RR-12 at Robinson until the 32nd refueling outage, which is scheduled to commence in September of 2020.

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

Principal Contributor: Jay Collins

Date: June 25, 2018

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2 – RELIEF REQUEST (RR)-12 REGARDING PROPOSED ALTERNATIVE TO ASME CODE CASE N-729-4 EXAMINATION FREQUENCY REQUIREMENTS (EPID L-2017-LLR-0089) DATED JUNE 25, 2018

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