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Attn: Document Control Desk
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/RENEWED LICENSE NO. DPR-23

RELIEF REQUEST (RR)-11 FOR RELIEF FROM VOLUMETRIC/SURFACE EXAMINATION FREQUENCY REQUIREMENTS OF ASME CODE CASE N-729-1

Pursuant to 10 CFR 50.55a(a)(3)(i), Duke Energy Progress, Inc., hereby requests a relief from performing the required volumetric/surface examinations for the H. B. Robinson Steam Electric Plant (RNP), Unit No. 2 reactor vessel closure head (RVCH) components identified in ASME Code, Section XI, Code Case N-729-1. The details and justification for this request are provided by the enclosure to this letter.

Duke Energy Progress, Inc. requests approval of the enclosed relief request by January 2015 to support the Spring 2015 refueling outage.

This letter contains no new Regulatory Commitments.

If you have any questions concerning this matter, please contact Mr. Richard Hightower, Manager – Regulatory Affairs at (843) 857-1329.

Sincerely,

Sharon W. Peavyhouse

Sharon W. Peavyhouse
Director – Nuc Org Effectiveness

SWP/cac

Enclosure: H. B. Robinson Steam Electric Plant, Unit No. 2, Relief Request (RR)-11 for Relief from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1

cc: Mr. V. M. McCree, NRC Region II
Ms. Martha Barillas, NRC Project Manager, NRR
NRC Resident Inspector

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H. B. Robinson Steam Electric Plant, Unit No. 2, Relief Request (RR)-11 for Relief from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1

1. **American Society of Mechanical Engineers (ASME) Code Component(s) Affected**

The affected components are ASME Class 1 Pressurized Water Reactor (PWR) Reactor Vessel Closure Head (RVCH) nozzles and partial-penetration welds fabricated with primary water stress corrosion cracking (PWSCC) resistant materials. H. B. Robinson Steam Electric Plant (RNP) Unit 2 penetration tubes and vent pipe are fabricated from Alloy 690 with alloy 52/152 attachment welds.

2. **Applicable Code Edition and Addenda**

The 5th inservice inspection (ISI) interval Code of record for RNP Unit 2 is the 2007 Edition with 2008 Addenda of ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

3. **Applicable Code Requirement**

The Code of Federal Regulations (CFR) 10CFR50.55a(g)(6)(ii)(D)(1), requires (in part):

All licensees of pressurized water reactors shall augment their ISI program with ASME Code Case N-729-1 (Ref. 1) subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of this section. Licensees of existing operating reactors as of September 10, 2008 shall implement their augmented ISI program by December 31, 2008.

10CFR50.55a(g)(6)(ii)(D)(3) conditions ASME Code Case N-729-1 by stating:

Instead of the specified 'examination method' requirements for volumetric and surface examinations in Note 6 of Table 1 of Code Case N-729-1, the licensee shall perform volumetric and/or surface examination of essentially 100 percent of the required volume or equivalent surfaces of the nozzle tube, as identified by Figure 2 of ASME Code Case N-729-1. A demonstrated volumetric or surface leak path assessment, through all J-groove welds shall be performed. If a surface examination is being substituted for a volumetric examination on a portion of a penetration nozzle that is below the toe of the J-groove weld [Point E on Figure 2 of ASME Code Case N-729-1], the surface examination shall be of the inside and outside wetted surface of the penetration nozzle not examined volumetrically. ASME Code Case N-729-1 specifies that the reactor vessel upper head components shall be examined on a frequency in accordance with Table 1 of this code case.

4. **Reason for Request**

Treatment of Alloy 690 RPV Closure Heads in Code Case N-729-1 was intended to be conservative and subject to reassessment once additional laboratory data and plant experience on the performance of Alloy 690 and Alloy 52/152 weld metals become available. Using plant and laboratory data, Electric Power Research Institute (EPRI) document Materials Reliability Program (MRP) - 375 was developed to support a technically based volumetric / surface re-examination interval using appropriate analytical tools. This technical basis demonstrates that the re-examination interval can be extended to the requested interval length while maintaining an acceptable level of quality and safety. Duke Energy is requesting approval of this alternative to allow the use of the ISI interval extension for the affected RNP Unit 2 component.

The expedited nature of this request is the result of Duke Energy only just becoming aware of recent industry requests for relief from Code Case N-729-1 by plants required to perform the inspections, and the Company's interest in significant dose reduction during the Spring 2015 refueling outage.

5. Proposed Alternative and Basis for Use

Proposed Alternative

Pursuant to 10CFR 50.55a (a)(3)(i), Duke Energy requests an alternative from performing the required volumetric/surface examinations for the RNP RVCH components identified above at the frequency prescribed in ASME Code, Section XI, Code Case N-729-1. Specifically, Duke Energy requests to extend the frequency of the volumetric/surface examination of the RNP RVCH of Table 1, Item B4.40 of ASME Code Case N-729-1 for approximately 3 years beyond the one inspection interval (nominally 10 calendar years) from installation of the RNP replacement RVCH. This request would extend the volumetric/surface examination to the 31st refueling outage which is scheduled to commence in September of 2018. No alternative examination processes are proposed to those required by ASME Code Case N-729-1, as conditioned by 10CFR50.55a(g)(6)(ii)(D). The visual examinations and acceptance criteria as required by Item B4.30 of Table 1 of ASME Code Case N-729-1 are not affected by this request and will continue to be performed on a frequency of every 3RD refueling outage or 5 calendar years, whichever is less.

Basis for Use

The original RNP RVCH, which was manufactured with Alloys 600/82/182 materials, was replaced with a new RVCH using Alloys 690/52/152 material during the refueling outage that returned to operation in October 2005. In accordance with Table 1 of ASME Code Case N-729-1, Item B4.40, as conditioned by 10CFR50.55a(g)(6)(ii)(D)(3), RNP will be required to perform a volumetric and/or surface examination of essentially 100% of the RVCH by the end of 2015.

The basis for the inspection frequency for ASME Code Case N-729-1 comes, in part, from the analysis performed in EPRI Materials Reliability Program (MRP)-111 (Ref. 2) which was summarized in the safety assessment for RVCHs in EPRI MRP-110 (Ref. 3). The material improvement factor for Primary Water Stress Corrosion Cracking (PWSCC) of Alloys 690/52/152 materials over that of mill annealed Alloys 600/82/182 was shown by this report to be in the order of 26 or greater.

Additional Evaluations Performed under EPRI MRP-375

Further evaluations were performed to demonstrate the resistance of Alloys 690/52/152 to PWSCC under a recent EPRI MRP initiative provided in EPRI MRP-375 (Ref. 4). This report presents both deterministic and probabilistic evaluations that assess the improved PWSCC resistance of Alloys 690/52/152.

Operating experience to date for replacement and repaired components using Alloys 690/52/152 has shown a proven record of resistance to PWSCC during numerous examinations in the 20+ years of its application. This includes steam generators, pressurizers, and RVCHs.

In particular, at the completion of the spring 2014 refueling outage season, Alloys 690/52/152 operating experience includes inservice volumetric/surface examinations performed on thirteen of the 40 plant replacement RVCHs in the US in accordance with ASME Code Case N-729-1.

In France in 2013, a second 10 year nondestructive examination (NDE) inspection was performed on one of the first reactor vessel (RV) heads to be replaced with alloy 690/52/152 material. There were no reports of PWSCC having been detected after approximately 20 years of service.

The evaluation performed in MRP-375 considers a simple Factor of Improvement (FOI) approach applied in a conservative manner to model the increased resistance of Alloy 690 compared to Alloy 600 at equivalent temperature and stress conditions. Even though base metal and welding variability of test data exist (i.e. heat affected zones, weld dilution zones, etc.), relative, but conservative, FOIs were estimated for the material improvements of Alloys 690/52/152 materials using an extensive database of test data. Results for both crack initiation and crack growth conclude a higher resistance to PWSCC for Alloy 690 base material and Alloy 52/152 weld materials. EPRI MRP-375, Figures 3-2, 3-4, and 3-6 provide crack growth data for Alloy 690/52/152 materials and heat affected zones with represented curves plotting FOIs of 1, 5, 10, and 20. A FOI of 20 bounds most of the data plotted, however, a FOI of 10 or less bounds all of the data.

EPRI MRP-375, Table 3-6 provides a summary of crack growth rate (CGR) and crack initiation data. For crack initiation, FOIs reported although significant, are conservative because, in many cases, crack initiation of Alloys 690/52/152 was not observed during testing; instead, the initiation time was assumed to be equivalent to the test duration. Additionally, many of the Alloy 690 crack growth rate tests were performed on specimens with considerable amounts of cold work (up to 40%), which is known to accelerate CGRs to rates that are not representative of cold work levels applicable to reactor vessel head penetrations.

EPRI MRP-375 then performed a combination of deterministic and probabilistic evaluations to establish a reasonable inspection interval for Alloy 690 RVCHs. The deterministic technical basis applies industry-standard crack growth calculation procedures to predict time to certain adverse conditions under various conservative assumptions. A probabilistic evaluation is then applied to make predictions for leakage and ejection risk generally using best-estimate inputs and assumptions, with uncertainties treated using statistical distributions.

The deterministic crack growth evaluation provides a precursor to the probabilistic evaluation to directly illustrate the relationship between the improved PWSCC growth resistance of Alloys 690/52/152 and the time to certain adverse conditions. These evaluations apply conservative CGR predictions and the assumption of an existing flaw (which is replaced with a PWSCC initiation model for probabilistic evaluation). The evaluations provide a reasonable lower bound on the time to adverse conditions, from which a *conservative* inspection interval may be recommended. This evaluation draws from various EPRI MRP and industry documents which evaluate, for Alloys 600/82/182, the time from a detectable flaw being created to leakage occurring and from a leaking flaw to the time that net section collapse (nozzle ejection) would be predicted to occur.

Applying a conservative crack growth FOI of 20 to circumferential and inside diameter (ID) axial cracking and of 10 to outside diameter (OD) axial cracking for Alloys 690/52/152 versus Alloys 600 and 182, the results show that more than 20 years is required for leakage to occur and that more than 120 years would be required to reach the critical crack size subsequent to leakage. The probabilistic model in EPRI MRP-375 was developed to predict PWSCC degradation and its associated risks in RVCHs.

The model utilized in this probabilistic evaluation is modified from the model presented in Appendix B of EPRI MRP-335, Rev. 1 (Ref. 5) that evaluated surface stress improvement of Alloy 600 RVCHs for surface stress improvement. The integrated probabilistic model in EPRI MRP-375 includes submodels for simulating component and crack stress conditions, PWSCC initiation, PWSCC growth, and flaw examination. The submodels for crack initiation and growth prediction for Alloy 600 reactor pressure vessel head penetration nozzles (RPVHPNs) in MRP-335, Rev. 1 were adapted for Alloy 690 RVCHs by applying FOIs to account for its superior PWSCC resistance. The probabilistic calculations are based on a Monte Carlo simulation model including PWSCC initiation, crack growth, and flaw detection via ultrasonic testing. The average leakage frequency and average ejection frequency were determined using conservative FOI assumptions. The results show that using only modest FOIs for Alloys 690/52/152 RVCHs, the potential for developing a safety significant flaw (risk of nozzle ejection) is acceptably small for a volumetric/surface examination period of 20 years.

The evaluations performed in EPRI MRP-375 were prepared to bound all PWR replacement RVCH designs that are manufactured using Alloy 690 base material and Alloy 52/152 weld materials. The evaluations assume a bounding continuously operating RVCH temperature of 613°F and a relatively large number of RVCH penetrations (89).

While Duke Energy is not requesting NRC review and approval of EPRI MRP-375 to approve this request for alternative, the insights gained in this technical report help substantiate the limited extension duration being requested for RNP of approximately 3 years beyond the 10 year examination frequency established in ASME Code Case N-729-1. In particular, the tabulation of CGR data for Alloys 690/52/152 (Section 3 of EPRI MRP-375) and review of inspection experience for Alloys 690/52/152 plant components (Section 2 of EPRI MRP-375) are sufficient to demonstrate the acceptability of the limited extension duration being requested. This request is not dependent on the more detailed probabilistic calculations presented in Section 4 of EPRI MRP-375.

RNP Unit 2 RVCH Design and Operation

The analysis performed by EPRI MRP-375 bounds the design and operation of the RNP replacement RVCH. The RVCH contains forty-seven (47) nozzle penetrations of which forty-five (45) are used for control element drive mechanisms (CEDMs), and two (2) small diameter penetrations near the center of the RVCH are used for the Reactor Head Vent (RHV) and Reactor Vessel Level Indication System (RVLIS). The Replacement RVCH was manufactured by Mitsubishi and placed in service in October 2005. The replacement RVCH was manufactured as a single forging which eliminated the center disc and flange circumferential weld in the original RNP RVCH. The replacement RVCH is fabricated from SA-508, Grade 3, Class 1 steel and clad with an initial layer of 309 L stainless steel followed by subsequent layers of 308 L stainless steel.

The nozzle housing penetrations on the replacement RVCH are fabricated from Inconel SB-167 (Alloy 690) UNS N06690 and the vent pipe was made from SB-167 (Alloy 690) and SA-312 Type 316.The nozzle J-groove welds utilized ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152) weld materials.

A preservice volumetric examination of the RNP replacement RVCH J-groove welded CEDM, RHV and RVLIS nozzles was performed by Westinghouse prior to installation. The volumetric examinations included scanning the nozzles to the fullest extent possible, from the end of the nozzle to a minimum of 2 inches above the root of the J-groove weld on the uphill side. There were no ultrasonic examination (UT) responses indicative of planar flaws identified during the volumetric examinations. Additionally, a preservice eddy current examination of the CEDM, RHV and RVLIS nozzle welds was performed. There were no responses indicative of planar flaws identified during the eddy current examinations.

A bare metal visual examination was performed in 2010 of the RNP replacement RVCH in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. This visual examination was performed by visual examination (VT-2) qualified examiners on the outer surface of the RVCH including the annulus area of the penetration nozzles. This examination did not reveal any surface or nozzle penetration boric acid that would be indicative of nozzle leakage. This examination will be performed again in the upcoming 29th refueling outage scheduled to commence in May 2015.

The EPRI MRP-375 analyses assume a reactor vessel head operating temperature of 613°F to bound the known RV head temperatures of all PWRs currently operating. The nominal operating hot leg temperature for RNP is 604.1°F. Core bypass flow is expected to reduce the upper head temperature by approximately 4.35°F, which would result in an average RVCH temperature of approximately 599.75°F. Based on this, the RNP RVCH average operating temperature (which is the measure of temperature relevant to potential PWSCC degradation) is bounded by the EPRI MRP-375 evaluation results, which assumes 613°F for its main deterministic and probabilistic calculations.

FOI Implied by Inspection Period

Duke Energy has also assessed the representative Alloy 690/52/152 FOI for the requested RNP extension period for comparison with the full set of laboratory CGR data. ASME Code Case N-729-1 is based upon conclusions reached that a head with Alloy 600 nozzles and operating at a temperature of 605°F is safe to operate up to 2 years (one 24 month operating cycle) between volumetric/surface examinations. The same period for Alloy 690 RVCHs in N-729-1 is 10 years which represents a factor of 5 over the Alloy 600 RVCHs. A simple extension of that improvement factor to 13 years would be a factor of 6.5 for the proposed period between volumetric/surface examinations for RNP.

However, the RVCH operating temperature assumed in the technical basis for heads with Alloy 600 nozzles (References 3, 6, & 7) for ASME Code Case N-729-1 was 605.0 F, compared to an assumed operating temperature of 599.75°F for RNP. Code Case N-729-1 addresses the effect of differences in operating temperature on the required volumetric/surface re-examination interval for heads with Alloy 600 nozzles on the basis of the re-inspection years (RIY) parameter. The RIY parameter adjusts the effective full power years (EFPYs) of operation between inspections for the effect of head operating temperature using the thermal activation energy appropriate to PWSCC crack growth.

For heads with Alloy 600 nozzles, ASME Code Case N-729-1 as conditioned by 10CFR50.55a limits the interval between subsequent volumetric/surface inspections to RIY = 2.25. The RIY parameter, which is referenced to a head temperature of 600°F, limits the time available for potential crack growth between inspections. As discussed in the technical basis documents for heads with Alloy 600 nozzles, effective time for crack growth is the principal basis for setting the appropriate reexamination interval to detect any PWSCC in a timely fashion. U.S. PWR inspection experience for heads with Alloy 600 nozzles has confirmed that the RIY = 2.25 interval results in a suitably conservative inspection program.

There have been no reports of nozzle leakage or of safety significant circumferential cracking subsequent to the time that the Alloy 600 nozzles in a head were first examined by non-visual inservice non-destructive examination for plants conforming to the 2.25 RIY Interval (References 8 & 9).

The representative RNP RVCH operating temperatures of 599.75°F would result in an RIY temperature adjustment factor of 0.994 (versus the reference temperature of 600.0 F) using the activation energy of 31 kcal/mol for crack growth of ASME Code Case N-729-1. Laboratory PWSCC crack growth rate testing for Alloy 690 wrought material by multiple investigators (References 10, 11, & 12) has shown thermal activation energy values comparable to the standard activation energy applied to model growth of Alloys 600/82/182 (31 kcal/mol or 130 kJ/mol). Thus, it is appropriate to apply this standard activation energy for modeling crack growth of Alloy 690/52/152 plant components. Conservatively assuming that the EFPYs of operation accumulated at RNP since RVCH replacement is equal to the calendar years since replacement, the RIY for the requested extended period at RNP would be (0.994) x (13 years) = 12.92 RIY. The FOI implied by this RIY value for RNP is (12.92)/(2.25) = 5.7 FOI. Considering the statistical compilation of data provided in Figures 3-2, 3-4, and 3-6 of EPRI MRP-375, this factor of improvement is conservatively less than the FOI of 10 that bounds the crack growth rate data presented. Furthermore, as discussed in Sections 2 and 3 of EPRI MRP-375, PWR plant experience and laboratory testing have demonstrated a large improvement in resistance to PWSCC initiation of Alloys 690/52/152 in comparison to that for Alloys 600/82/182. Hence, the demonstrated improvements in PWSCC initiation and growth confirm on a conservative basis the acceptability of the limited requested period of extension.

Conclusions

Duke Energy believes that the Alloy 690 nozzle base and Alloy 52/152 weld materials used in the RNP replacement RVCH provide for a clearly superior reactor coolant system pressure boundary where the potential for PWSCC has been shown by analysis and by years of positive industry experience to be remote. This is further supported by visual examination of the RNP RVCH in 2010 and the volumetric examinations performed by other Westinghouse designed plants during their nominal 10-year examination under similar operating conditions which did not reveal PWSCC.

The FOI implied by the requested extension period represents a level of reduction in PWSCC crack growth rate versus that for Alloys 600/82/182 that is completely bounded on a statistical basis by the laboratory data compiled in EPRI MRP-375. Given the lack of PWSCC detected to date in any PWR plant applications of Alloys 690/52/152, the simple FOI assessment clearly supports the limited requested period of extension.

Therefore, the RNP RVCH FOI corresponding to the requested period of extension to perform a volumetric/ surface examination provides an acceptable level of quality and safety in accordance with 10CFR50.55a(a)(3)(i).

6. Duration of Proposed Alternative

The proposed alternative is requested for the duration up to and including the 31st RNP refueling outage that is schedule to commence in September of 2018 and which will occur in the fifth ten-year ISI inspection interval which began July 22, 2012 and ends July 30, 2021.

7. Precedents

ML14118A477 - Request for Alternative from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1, Arkansas Nuclear One, Unit 1 - Currently under NRC review.

ML14206A939 - Request for Alternative from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1, St Lucie Unit 1 - Currently under NRC review.

8. References

1 ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," Approved March 28, 2006.
2 EPRI MRP-111, "Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors," Report No. 1009801, March 2004 (ML041680546).
3 EPRI MRP-110, "Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants," Report No. 1009807, April 2004 (ML041680506).
4 EPRI MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles", Report No. 3002002441, February 2014 (publically available at www.epri.com)
5 EPRI MRP-335 (Rev. 1), "Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement," Report No. 3002000073, January 2013.
6 EPRI MRP-117, "Inspection Plan for Reactor Vessel Closure Head Penetrations in U.S. PWR Plants," Report No. 1007830, December 2004 (ML043570129).
7 EPRI MRP-105, "Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking," Report No. 1007834, April 2004 (ML041680489).
8 EPRI MRP Letter 2011-034, "Tcold RV Closure Head Nozzle Inspection Impact Assessment," dated December 21, 2011 (ML12009A042)
9 G. White, V. Moroney, and C. Harrington, "PWR Reactor Vessel Top Head Alloy 600 CRDM Nozzle Inspection Experience," presented at EPRI International BWR and PWR Material Reliability Conference, National Harbor, Maryland, July 19, 2012.
10 U.S. NRC, "Stress Corrosion Cracking in Nickel-Base Alloys 690 and 152 Weld in Simulated PWR Environment- 2009," NUREG/CR-7137, ANL-10/36, published June 2012 (ML1 2199A41 5).
11 EPRI MRP-237 (Rev. 2), "Resistance of Alloys 690, 152, and 52 to Primary Water Stress Corrosion Cracking: Summary of Findings Between 2008 and 2012 from Completed and Ongoing Test Programs," Report No. 3002000190, April 2013 (publically available at www.epri.com)

12 M. B. Toloczko, M. J. Olszta, and S. M. Bruemmer, "One Dimensional Cold Rolling Effects on Stress Corrosion Crack Growth in Alloy 690 Tubing and Plate Materials," *15th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors,* TMS (The Minerals, Metals & Materials Society)