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10 CFR 50.55(a)

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

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

SUBJECT: Relief Requested in Accordance with 10 CFR 50.55a(z)(1) for Volumetric or Surface Examination of Code Case N-729-4 Examinations of the Reactor Pressure Vessel Upper Head

Ladies and Gentlemen:

In accordance with 10 CFR 50.55a(z)(1), Duke Energy Progress, LLC (Duke Energy) requests approval for H.B. Robinson Steam Electric Plant Unit 2 (Robinson Unit 2) to extend the examination of Reactor Vessel Closure Head (RVCH) nozzles and partial-penetration welds from the refueling outage scheduled to commence in September 2020 to the refueling outage scheduled to commence in September 2024. The proposed extension would equate to an examination interval of 19 years. Duke Energy is requesting the extension on the basis that the alternative provides an acceptable level of quality and safety.

Duke Energy is requesting review and approval of the Relief Request by August 1, 2020 to support the use of the proposed alternative.

Relief Request RA-19-0106 is provided as an enclosure to this letter.

This document contains no new Regulatory Commitments.

Should you have any questions concerning this letter, or require additional information, please contact Art Zaremba, Director – Nuclear Fleet Licensing, at 980-373-2062.

Sincerely,

Kevin Ellis – Manager – Nuclear Support Services, H.B. Robinson Steam Electric Plant, Unit 2.

Enclosure 1: Relief Request RA-19-0106

cc: L. Dudes, NRC Regional Administrator, NRC, Region II
J. Hammon, NRC Senior Resident Inspector, HBRSEP, Unit No. 2
N. Jordan, NRC Project Manager, NRR

Enclosure 1

Duke Energy Progress, LLC

H. B. Robinson Steam Electric Plant, Unit 2

Relief Request Number RA-19-0106

Relief Requested in Accordance with 10 CFR 50.55a(z)(1) for Volumetric or Surface Examination of Code Case N-729-4 Examinations of the Reactor Pressure Vessel Upper Head

1.0 American Society of Mechanical Engineers (ASME) Code Component(s) Affected

1.1 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 material.

2.0 Applicable Code Edition and Addenda

2.1 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".

3.0 Applicable Code Requirement

3.1 Code of Federal Regulations (CFR) 10CFR50.55a(g)(6)(ii)(D) delineates the Augmented ISI requirements for Reactor Vessel Head Inspections. 10CFR50.55a(g)(6)(ii)(D)(1), requires:

> "Holders of operating licenses or combined licenses for pressurized-water reactors as of or after August 17, 2017 shall implement the requirements of ASME BPV Code Case N–729–4 instead of ASME BPV Code Case N–729–1, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) of this section, by the first refueling outage starting after August 17, 2017."

3.2 ASME Code Case N-729-4, -2410 specifies the Inspection Program requirements for the reactor vessel upper head and its penetrations (nozzles and partial-penetration welds). The basic inspection requirements of Code Case N-729-4, -2410 as contained within Table 1 for partial-penetration welded Alloy 690 head penetration nozzles are as follows:

3.2.1 Item Number B4.40: Volumetric or surface examination of all nozzles, not to exceed one inspection interval (nominally 10 calendar years) provided that flaws attributed to primary water stress corrosion cracking (PWSCC) have not been identified.

3.2.2 Item Number B4.30: Direct visual examination (VE) of the outer surface of the head for evidence of leakage every third refueling outage or 5 calendar years, whichever is less.

3.3 The volumetric and/or surface examination frequency for Item Number B4.40 required in ASME Code Case N-729-4 is identical to that of Code Case N-729-1. The previous NRC conditions on Code Case N-729-1 and the current NRC conditions on Code Case N-729-4 in 10CFR50.55a(g)(6)(ii)(D) do not affect the examination frequency required for Item Number B4.40.

4.0 Reason for Request

4.1 The treatment of Alloy 690 RPV Closure Heads in Code Case Versions N-729 through N-729-4 (Reference 8.1) 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 became available. Electric Power Research Institute (EPRI) generated MRP-375 (Materials Reliability Program) (Reference 8.4), "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (EPRI 3002002441)" using plant and laboratory data. MRP-375 was developed to support a technically based volumetric or surface examination frequency using appropriate analytical tools. This technical basis demonstrates that the examination frequency can be extended to 20 years while maintaining an acceptable level of quality and safety. Additionally, new rulemaking (Reference 8.7) is expected to be published in late 2019 which will revise 10 CFR 50.55a to incorporate by reference ASME BPV Code Case N-729-6. The examination frequency delineated in Code Case N-729-6 for Item Number B4.40 has been extended from no more than one inspection interval (nominally 10 calendar years) to no more than 2 inspection intervals (20 calendar years).

4.2 The NRC previously approved deferral of the volumetric or surface inspection of the RNP Unit 2 head nozzles until the refueling outage that commenced in September of 2018 via RR-11 (Reference 8.2), and again until the refueling outage that is scheduled to commence in September of 2020 via RR-12 (Reference 8.3). These deferrals were intended to provide sufficient time for the NRC to review and accept the conclusions reached in MRP-375 (Reference 8.4).

4.3 Duke Energy is requesting approval of deferral beyond the previously approved extensions of the nominal 10-year examination frequency for Item B4.40 of Table 1 of Code Cases N-729-1 and N-729-4 until the refueling outage scheduled to commence in the fall of 2024. This extension would align with the new examination frequency required in Code Case N-729-6 (Reference 8.7) of 20 years for Item Number B4.40.

5.0 Proposed Alternative and Basis for Use

5.1 Pursuant to 10 CFR 50.55a(z)(1), Duke Energy requests an alternative from performing the required volumetric or surface examinations for the RNP RVCH components identified above at the frequency prescribed in ASME Code Case N-729-4. Specifically, Duke Energy requests to extend the frequency of the volumetric or surface examination of the RNP RVCH of Table 1, Item B4.40 of ASME Code Case N-729-4 to the 34th refueling outage, which is scheduled to commence in the fall of 2024. The proposed extension would equate to an examination interval of 19 years.

5.2 On March 3, 2016, the ASME approved the sixth revision of ASME BPV Code Case N-729-6. This revision includes changes that were deemed necessary by the

ASME to supersede the previous requirements under historical revisions of Code Case N-729 to establish an effective long-term inspection program for the RPV upper head penetration nozzles and associated welds in PWRs. One major change in the latest revisions is extending PWSCC resistant RPV upper head nozzle and partial-penetration weld inspection frequency from every 10 years to every 20 years. The extension requested by Duke Energy for RNP Unit 2 would align with the examination frequency delineated in Code Case N-729-6 of every 20 years. No alternative examination processes are proposed to those required by ASME Code Case N-729-4, 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-729-4 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D)(3), are not affected by this request and will continue to be performed as required.

5.3 As documented in Section 7.0 (Related Industry Relief Requests), NRC has approved similar extensions for other replacement RVCHs of more than 5 years beyond the nominal 10 years required by ASME Code Case N-729-4 to align with scheduled refueling outages. Duke Energy requests the total time deferral of approximately 10 years for RNP Unit 2 to align with the frequency required in ASME BPV Code Case N-729-6 which is scheduled to be published in rulemaking that should be issued in late 2019.

5.4 The original RNP RVCH, which was manufactured with Alloy 600/82/182 materials, was replaced with a new RVCH using Alloy 690/52/152 materials during the refueling outage that returned to operation in October 2005. In accordance with NRC approval of RNP Unit 2 Relief Request (RR-12), RNP Unit 2 will be required to perform a volumetric and/or surface examination of all nozzles during the 32nd refueling outage, which is scheduled to commence in September of 2020.

5.5 Additionally, evaluations have been performed to demonstrate the resistance of Alloys 690/52/152 to PWSCC under a recent EPRI MRP initiative provided in EPRI MRP-375 (Reference 8.4). This report presents both deterministic and probabilistic evaluations that assess the improved PWSCC resistance of Alloys 690/52/152.

5.6 RNP Unit 2 Design and Operation

5.6.1 The analysis performed by EPRI MRP-375 bounds the design and operation of the RNP Unit 2 replacement RVCH. The RVCH contains forty-seven (47) nozzle penetrations of which forty-five (45) are used for control rod drive mechanisms (CRDMs), 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.

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

5.6.3 A preservice volumetric examination of the RNP Unit 2 replacement RVCH J-groove welded CRDM, 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 two inches above the root of the J-groove weld on the uphill side. There were no ultrasonic examination responses indicative of planar flaws identified during the volumetric examinations. Additionally, a preservice eddy current examination of the CRDM, RHV and RVLIS nozzle welds was performed. There were no responses indicative of planar flaws identified during the eddy current examinations.

5.6.4 Bare metal visual examinations were performed in 2010 and 2015 in accordance with ASME Code Case N-729-1, Table 1, Item Number B4.30; and again in 2018 in accordance with ASME Code Case N-729-4, Table 1, Item Number B4.30. These visual examinations (VE) were performed by visual examination (VT-2) qualified examiners on the outer surface of the RVCH including the annulus area of the penetration nozzles. These examinations did not reveal any surface or nozzle penetration boric acid that would be indicative of nozzle leakage.

5.6.5 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 Unit 2 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 evaluation results in EPRI MRP-375, which assumes 613°F for its main deterministic and probabilistic calculations.

5.7 Conclusions

5.7.1 Duke Energy believes that the Alloy 690 nozzle base and Alloy 52/152 weld materials used in the RNP Unit 2 replacement RVCH provide for a 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, 2015, and 2018 all resulting in no relevant indications, and the volumetric examinations performed by other Westinghouse designed plants during their nominal 10-year volumetric or surface examination under similar operating conditions which did not reveal PWSCC.

5.7.3 The proposed revised examination frequency will continue to provide reasonable assurance of structural integrity. Additionally, the proposed frequency aligns with the examination interval delineated in Code Case N-729-6 which is projected to be approved for use in the new rule making scheduled to be issued in late 2019. For the reasons noted above, it is requested that the NRC authorize this proposed alternative in accordance with 10CFR50.55a(z)(1) as the alternative provides an acceptable level of quality and safety.

6.0 Duration of Proposed Alternative

6.1 The proposed alternative is requested for the duration up to and including the 34th RNP Unit 2 refueling outage that is scheduled to commence in the fall of 2024.

7.0 Related Industry Relief Requests

7.1 There have been submittals from multiple plants requesting an alternative from the nominal 10-year interval of ASME Code Cases N-729-1 and N-729-4 for volumetric or surface examinations of heads with Alloy 690 nozzles. The prior RNP Unit 2 requests for relief and requests from other plants, including the associated status at the time of submittal of this request, are shown below in Table 7-1. Alternative intervals greater than 15 years have previously been granted to align with scheduled refueling outages. The approved Calvert Cliffs Units 1 & 2 alternative (noted in the table below) permitted an inspection interval not to exceed 16 years to align with scheduled refueling outages. Furthermore, alternatives have been approved at three sites that further extend the inspection interval from an initial approved alternative to a total interval of up to 15.5 years at Arkansas Nuclear One 1, Beaver Valley 1, and St. Lucie 1.

Table 7-1

Plant	NRC ADAMS Accession No.				Status
	Relief Request	Request for Additional Information (RAI)	RAI Response	NRC Safety Evaluation	
Arkansas Nuclear One, Unit 1 (2.5-year Extension)	ML14118A477	ML14258A020	ML14275A460	ML14330A207	Approved
Arkansas Nuclear One, Unit 1 (5.5-year Extension)	ML16173A297	None	None	ML17018A283	Approved
Beaver Valley, Unit 1 (2-year Extension)	ML14290A140	None	None	ML14363A409	Approved
Beaver Valley, Unit 1 (5-year Extension)	ML17044A440	None	None	ML17222A162	Approved
Calvert Cliffs, Units 1 & 2	ML15201A067	None	None	ML15327A367	Approved
Comanche Peak, Unit 1	ML15120A038	None	None	ML15259A004	Approved
D.C. Cook, Units 1 & 2	ML15023A038	None	None	ML15156A906	Approved
J.M. Farley, Unit 2	ML15111A387	None	None	ML15104A192	Approved
North Anna, Unit 2	ML14283A044	None	None	ML15091A687	Approved
Prairie Island, Units 1 and 2	ML14258A124	ML15030A008	ML15036A252	ML15125A361	Approved
Palo Verde, Units 1, 2, and 3	ML17299B333	None	None	ML18040A331	Approved
H. B. Robinson, Unit 2	ML14251A014	ML14294A587	ML14325A693	ML15021A354	Approved
H. B. Robinson, Unit 2	ML17269A016	None	None	ML18163A412	Approved
Salem, Unit 1	ML15098A426	None	None	ML15349A956	Approved
St. Lucie, Unit 1 (3-year Extension)	ML14206A939	ML14251A222	ML14273A011	ML14339A163	Approved
St. Lucie, Unit 1 (5.5-year Extension)	ML17045A357	None	None	ML17219A174	Approved
St. Lucie, Unit 2	ML16076A431	None	None	ML16292A761	Approved

8.0 References

8.1 ASME Code Case N-729-4, “Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1,” Approved June 22, 2012

8.2 U.S. NRC, “H. B. Robinson Steam Electric Plant, Unit No. 2 - Relief Request (RR-11) - Relief from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1 (TAC No. MF4801)” August 27, 2014 (ML14251A014).

8.3 U.S. NRC, “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)” June 25, 2018 (ML18163A412)

8.4 EPRI MRP-375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," Report No. 3002002441, February 2014

8.5 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).

8.6 EPRI MRP-110, "Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants," Report No.1009807, November 2004 (ML041680506).

8.7 Federal Register/Vol. 83, No. 218/Friday, November 9, 2018/Proposed Rules, FR Doc. 2018-24076