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

January 28, 2015

Mr. Eric A. Larson, Site Vice President
FirstEnergy Nuclear Operating Company
Beaver Valley Power Station
Mail Stop A-BV-SEB1
P.O. Box 4, Route 168
Shippingport, PA 15077

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 - RELIEF REQUEST NO. 1TYP-4-RV-04 REGARDING THE EXAMINATION REQUIREMENTS OF CODE CASE N-729-1 (TAC NO. MF5049)

Dear Mr. Larson:

By letter dated October 17, 2014,[1] FirstEnergy Nuclear Operating Company (the licensee) submitted request 1TYP-4-RV-04 to the Nuclear Regulatory Commission (NRC). The licensee requested to use alternative requirements to the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code), Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," for Beaver Valley Power Station, Unit 1 (BVPS-1).

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i), (retitled paragraph 50.55a(z)(1) by 79 FR 65776, dated November 5, 2014), the licensee requested an alternative to performing the required volumetric and surface examinations for certain reactor pressure head components at the frequency prescribed in ASME Code Case N-729-1.

The NRC staff has reviewed the licensee's relief request and has determined that the requested alternative will provide an acceptable level of quality and safety, as documented in the enclosed safety evaluation. Therefore, the licensee's request for the use of the above stated alternative is authorized pursuant to 10 CFR 50.55a(z)(1) for the BVPS-1.

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

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[1] Agencywide Documents Access and Management System Accession No. ML14290A140.

If you have any questions, please contact the Beaver Valley Project Manager, Taylor A. Lamb, at (301) 415-7128 or Taylor.Lamb@nrc.gov.

Sincerely,

Meena Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure:
Safety Evaluation

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST REGARDING THE PROPOSED ALTERNATIVE TO ASME CODE CASE

N-729-1 EXAMINATION FREQUENCY REQUIREMENTS

FIRSTENERGY NUCLEAR OPERATING COMPANY

BEAVER VALLEY POWER STATION, UNIT 1

DOCKET NUMBER 50-334

1.0 INTRODUCTION

By letter dated October 17, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML14290A140), FirstEnergy Nuclear Operating Company (the licensee or FENOC) requested relief from the requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI, associated with the examination frequency requirements of Code Case N-729-1 for Beaver Valley Power Station, Unit No. 1 (BVPS-1).

Specifically, pursuant to Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a(a)(3)(i) (retitled paragraph 50.55a(z)(1) by 79 FR 65776, dated November 5, 2014), the licensee requested to use the proposed alternatives in Relief Request 1TYP-4-RV-04, to the examination frequency of ASME Code Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," on the basis that the alternative examination provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(g)(6)(ii) state that "the Commission may require the licensee to follow an augmented inservice inspection [ISI] program for systems and components for which the Commission deems that added assurance of structural reliability is necessary."

The regulations in 10 CFR 50.55a(g)(6)(ii)(D), state, in part, that "all licensees of pressurized water reactors shall augment their [ISI] program with ASME Code Case N 729-1, subject to conditions specified in paragraphs (g)(6)(ii)(D)(2) through (6) of this section . . ."

In this request, the licensee has requested relief from the examination frequency required by Code Case N-729-1 and has, therefore, also requested relief from 10 CFR 50.55a(g)(6)(ii)(D).

Enclosure

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 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.

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 Commission to authorize the proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 COMPONENTS AFFECTED

The affected components are ASME Class 1, Reactor Vessel Closure Head (RVCH) Penetration Nozzle numbers 1 through 62, which are fabricated from Inconel SB-167 (Alloy 690) UNS N06690. The nozzle J-groove welds are fabricated from UNS N06052 and UNS W86152, 52/152 weld materials. The original BVPS-1 RVCH penetration nozzles, which were manufactured with Alloy 600/82/182 materials, was replaced with a new RVCH using Alloy 690/52/152 material for the penetration nozzles during the refueling outage that returned to operation in April 2006.

3.2 INSERVICE INSPECTION INTERVAL

The licensee's current ISI is the fourth 10-year ISI, which started on April 1, 2008, and ends on March 31, 2018. The proposed duration of the alternative is requested for the duration up to and including the 25th BVPS-1 refueling outage that is scheduled to commence in April 2018.

3.3 ASME CODE OF RECORD

The ASME Section XI Code of Record for the current, fourth 10-year ISI interval at BVPS-1, is the 2001 Edition through the 2003 Addenda.

3.4 ASME CODE AND/OR REGULATORY REQUIREMENTS

Section 50.55a(g)(6)(ii)(D) of 10 CFR requires, in part, that licensees augment their ISI program in accordance with ASME Code Case N-729-1, subject to the conditions specified in paragraphs (2) through (6) of 10 CFR 50.55a(g)(6)(ii)(D). ASME Code Case N-729-1, Table 1, Inspection Item B4.40 requires volumetric/surface examination be performed within one inspection interval (nominally 10 calendar years) of its inservice date for a replaced RVCH. The required volumetric/surface examinations would thus have to be completed by early 2016 in order to fulfill the requirements of N-729-1.

3.5 PROPOSED ALTERNATIVE

The licensee proposes to delay the next required inspection for a period of approximately 2 years. The licensee proposes to accomplish the inspection in accordance with ASME Code

Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D) during refueling outage 25, which is scheduled for April 2018.

3.6 LICENSEE'S BASIS FOR USE OF THE PROPOSED ALTERNATIVE

The licensee's basis for use of the proposed alternative is based primarily on three topics of consideration. The first topic addresses the concept that the inspection interval in Code Case N-729-1 is based on primary water stress-corrosion cracking (PWSCC) crack growth rates for Alloy 600/82/182. The second topic addresses a bare metal visual examination that was conducted on the licensee's replacement RVCH in 2010. The third topic addresses a plant-specific factor of improvement analysis that was conducted by the licensee.

In addressing its first basis for use of the proposed alternative, the licensee asserts that the inspection intervals contained in ASME Code Case N-729-1 for alloy 600/82/182 are based on re-inspection years (RIY) equal to 2.25. This RIY value is based on PWSCC crack growth rates as defined in the 75th percentile curve contained in Materials Reliability Program (MRP) 55, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," and MRP 115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," both NRC approved documents. The licensee further asserts that the PWSCC crack growth rates of Alloy 690/52/152 are significantly lower than those of Alloy 600/82/182 and, therefore, merit a longer inspection interval. The licensee bases that assertion on: (1) the lack of cracking in other Alloy 690 components, such as steam generators and pressurizers, in the nearly 20 years that Alloy 690 has been in service in these components; (ii) the failure to observe cracking in inspections already performed in replacement heads (13 of 40 replacement heads in the United States have been examined, which includes heads that operate at higher temperatures than the head under consideration); (iii) the similarity of the inspected heads to the head under consideration regarding configuration, manufacturers, design and operating conditions; and (iv) laboratory test data for Alloy 690/52/152 as contained in MRP 375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles."

In addressing its second basis for use of the proposed alternative, the licensee stated that a bare metal visual examination was performed in 2010 on the BVPS-1 replacement RVCH in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. This visual examination was performed by 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 indications of nozzle leakage (e.g., boric acid deposits) on the surface or near a nozzle penetration. The licensee also indicated that this examination will be performed again in the upcoming refueling outage, which is scheduled to commence in April 2015. Also, the licensee stated that no alternative examination processes are proposed to those required by ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The visual (VT-2) 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 not to exceed every 5 calendar years.

In addressing its third basis for use of the proposed alternative, the licensee made a plant-specific calculation of the required factor of improvement in the crack growth rate of Alloy

690/52/152, as compared to the crack growth rate of Alloy 600/82/182. In making this calculation, the licensee used the actual temperature of the head and conservatively assumed that calendar years were equal to effective full-power years. Based on this calculation, the licensee determined that an improvement factor of 5.5 was required to meet the proposed and desired inspection interval of 12 calendar years. The licensee then proposed that, because the required factor of improvement (5.5) was smaller than the factor of improvement of 20, which bounded most of the MRP 375 data for Alloy 690/52/152, the use of a factor of improvement of 5.5 would not result in a reduction in safety and was, therefore, justified.

The licensee stated that their analysis showed significant margin to ensure that Alloy 690 nozzle base and Alloy 52/152 weld materials used in the BVPS-1 replacement RVCH provide for a 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. As such, the licensee found the technical basis sufficient to ensure public health and safety by extending the inspection frequency of the RVCH nozzle at BVPS-1 from a maximum of 10 years to a new maximum of 12 years.

3.7 NRC STAFF EVALUATION

In evaluating the technical sufficiency of the licensee's proposed alternative (i.e., a one-time extension of the volumetric/surface examination interval contained in ASME Code Case N-729-1 from 10 years to no longer than 12 years), the NRC staff considered each of the three aspects of the licensee's basis for use of the proposed alternative. The NRC staff found that the technical basis included by the licensee provided sufficient information for the NRC staff to review the proposed alternative.

Due to concerns about PWSCC, many PWR plants in the United States and overseas have replaced reactor vessel closure heads containing Alloy 600/182/82 nozzles with heads containing Alloy 690/152/52 nozzles. The inspection frequencies developed in Code Case N-729-1 for RVCH penetration nozzles using Alloy 600/182/82 were developed based, in part, on those material's crack growth rate equations, as documented in MRP 55 and MRP 115. The licensee's primary technical basis is to present crack growth rate data for the new, more crack resistant materials, Alloy 690/152/52, and demonstrate an improvement factor (IF) of these materials versus the older Alloy 600/82/182 materials. This IF would then provide the basis for the extension of the ISI frequency requested by the licensee in their proposed alternative.

In evaluating the licensee's first technical basis for use of the proposed alternative, the NRC staff notes that the licensee used MRP 375, "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles." This document, in part, summarizes numerous Alloy 690/152/52 crack growth rate data from various sources to develop IFs for the crack growth rate equations provided in MRP 55 and MRP 115. While the NRC staff finds the licensee's assertions and/or interpretations to be reasonable, MRP 375 is not an NRC approved document. As the NRC staff does not have sufficient time or resources to validate all of the data used by this document, the NRC staff does not consider it appropriate to use all of the data from this document to review the licensee's relief request. A more detailed review of the data provided in MRP 375 will be performed by an international group of experts as part of an Alloy 690 Expert Panel, which is currently scheduled to complete its review in the

2016-2017 timeframe. In the interim, the NRC staff review will rely upon Alloy 690/152/52 crack growth rate data from two NRC contractors: Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). This data is documented in a data summary report and can be found under ADAMS Accession Number ML14322A587. The NRC confirmatory research generally supports the contention that the crack growth rate of Alloy 690/52/152 is more crack resistant but differs from the MRP 375 data in some respects.

The PNNL and ANL data summary report includes crack growth rate data up to approximately 20 percent cold work based on the observation of local strains in welds and weld dilution zone data. However, the NRC staff did not consider the weld dilution zone data in its assessment. This is because the limited weld dilution zone data that is currently available has shown higher crack growth rates than are commonly observed for Alloy 690/152/52 material. The high crack growth rates in weld dilution zones may be due to the reduced chromium present in these areas. The NRC staff chose to exclude the weld dilution zone data from this analysis due to the limited number of data points available, the variability in results, and due to the limited area of continuous weld dilution for flaws to grow through. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a j-groove weld along the low Alloy steel head interface. It is not fully apparent to the NRC staff how accelerated crack growth in very small areas of the weld dilution zone would result in a significantly increased probability of leakage or component failure during a relatively short extension of the required inspection interval. Exclusion of these data may be reevaluated as additional data become available; a better understanding of the existing data is obtained; or if a longer extension of the inspection interval is requested. Therefore, the NRC staff finds that the impact of these weld dilution zone crack growth rates on the change in volumetric inspection frequency, as requested by the licensee's proposed alternative, is not considered to be relevant for this specific relief request.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds that the past bare metal visual examination on the head under consideration is a reasonable means to demonstrate the absence of leakage through the nozzle/J-groove weld, prior to the time the examination was conducted. The NRC staff also finds that performance of future bare metal visual examinations, in accordance with the code case, is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Finally, the NRC staff finds that the proposed alternative's frequency for bare metal visual examinations, in conjunction with the new frequency of volumetric examinations, is sufficient to provide reasonable assurance of the structural integrity of the RVCH.

In evaluating the licensee's third basis for use of the proposed alternative, the NRC staff found that the licensee's calculated improvement factor of 5.5, to support an extension of the ASME Code Case N-729-1 inspection frequency of 2.25 RIY to 12 calendar years, was found to be acceptable by NRC staff calculation. The NRC staff also found that the application of an IF of 5.5 to the 75th percentile curves in MRP 55 and 115 bounded essentially all of the NRC data included in the PNNL and ANL data summary report. Therefore, the NRC staff found that this analysis supports the concept that a volumetric inspection interval for the RVCH of not more than 12 calendar years does not pose a higher risk than that associated with an Alloy 600/182/82 RVCH inspected at intervals of 2.25 RIY. Hence, the NRC staff found the licensee's technical basis to be acceptable.

Therefore, based on the above evaluation, the NRC finds that the proposed alternative provides an acceptable level of quality and safety in accordance with 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 1TYP-4-RV-04, Revision 0, will provide an acceptable level of quality and safety for the examination frequency requirements of the RVCH. 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 one time use of 1TYP-4-RV-04 at Beaver Valley Power Station, Unit No.1, for the duration up to and including the 25th refueling outage that is scheduled to commence in April 2018.

All other requirements of the ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Steven Vitto

Date: January 28, 2015

If you have any questions, please contact the Beaver Valley Project Manager, Taylor A. Lamb, at (301) 415-7128 or Taylor.Lamb@nrc.gov.

Sincerely,

/RA/

Meena Khanna, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosure:
Safety Evaluation

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