From:	Ballard, Brent
To:	McCreary, Dave M
Cc:	Lashley, Phil H; Gonzalez, Hipo
Subject:	Request for Additional Information for Beaver Valley Unit 2 - 2R22 Steam Generator 180 Day Report (EPID L-2022-LRO-0064)
Date:	Thursday, August 25, 2022 1:00:00 PM
Attachments:	Final RAIs for Beaver Valley U2 180 Day Report.docx

Good afternoon Dave,

By letter dated May 5, 2022, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22126A089), Energy Harbor Nuclear Corp. (the licensee) submitted the Beaver Valley Unit 2 Refueling Outage 22 Steam Generator 180 Day Report, in accordance with Technical Specification 5.6.6.2. The report provides information required by the technical specifications that was obtain during the fall 2021 refueling outage inspections.

The NRC staff has determined that additional information is needed to complete its review of the reports. Attached is the NRC staff's request for additional information (RAI).

The NRC staff is requesting the licensee respond to the RAI by October 24, 2022. Please let me know if you have any questions.

Thank you,
Brent

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BEAVER VALLEY POWER STATION, UNIT 2

REQUEST FOR ADDITIONAL INFORMATION

UNIT #2 – 2R22 STEAM GENERATOR 180 DAY REPORT

DOCKET NO. 50-412

1.0 BACKGROUND

By letter dated May 5, 2022 (ML22126A089), Energy Harbor Nuclear Corp. (the licensee) submitted the Unit #2 – 2R22 Steam Generator 180 Day Report summarizing the results of the fall 2021 steam generator (SG) inspections performed at Beaver Valley Power Station, Unit 2 (Beaver Valley Unit 2). The inspections were performed during refueling outage 22 (2R22). Technical Specification (TS) Section 5.6.6.2 requires that a report be submitted within 180 days after the initial entry into hot shutdown (MODE 4) following completion of an inspection of the SGs performed in accordance with TS Section 5.5.5.2.

2.0 REGULATORY BASIS

The tubing of the steam generator constitutes more than half of the reactor coolant pressure boundary (RCPB). Design of the RCPB for purposes of structural and leakage integrity is a requirement under Title 10 of the Code of Federal Regulations Part 50 (10 CFR Part 50), Appendix A. Requirements governing the maintenance of steam generator tube integrity are in plant TS and in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). Specifically, Beaver Valley TS 5.5.5 requires a SG program be established and implemented to ensure that SG tube integrity is maintained. The SG program for Unit 2 shall include the provisions of TS 5.5.5.2, and SG tube inspection reports shall be submitted in accordance with TS 5.6.6.2. The TS 5.5.5.2 repair limits ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria (GDCs) 14, 15, 30, 31, and 32 of 10 CFR Part 50, Appendix A.

3.0 REQUEST FOR ADDITIONAL INFORMATION

To complete its review of the inspection, the U.S. Nuclear Regulatory Commission (NRC) staff requests the following additional information:

1. Table “2R22 Examination Techniques Used for Degradation Detection/Sizing” states that an “Internal Westinghouse Technique Qualification” was used for axial primary water stress corrosion cracking (PWSCC) at dented tube support plates (TSPs) less than 2 volts. It was noted that Electric Power Research Institute Examination Technique Specification Sheet 96012.1 was used in 2R21 (2020) for axial PWSCC at dented TSPs less than 2 volts (ML20287A373). Please provide additional information regarding why a different examination technique was used in 2R22 for axial PWSCC at dented TSPs less than 2 volts.

2. During 2R22, the +Point™ probe observed circumferential indications in four Alloy 800 nickel banded mechanical tubesheet sleeves (three installed in 2R20 (2018) and one installed in 2R21 (2020)) located at the lower tubesheet hard roll joint at the lower expansion

transition below the nickel band region. It was noted that one of the sleeves installed in 2R20 (2018) had a circumferential signal after sleeve installation and prior to being placed in service. While preliminary assessments suggested the signals could be circumferential outside diameter stress corrosion cracking (ODSCC) on the outer surface of the parent tube, it was determined that the circumferential signals may be fabrication related and not flaws because (1) flaws at this location are not likely due to the full-depth hard roll condition of the parent tube, (2) one circumferential signal had a precursor after sleeve installation and prior to being placed in service, and (3) the Ghent Version 2 probe did not observe signals like those observed by the +Point™ probe and did not detect any flaw-like signals. The affected tubes were proactively plugged during 2R22. Please discuss the following:

a. A comparison of the precursor eddy current signal in the one sleeve installed in 2R20 (2018) compared to the circumferential signal observed during 2R22.

b. Why the preliminary assessment suggested ODSCC on the outer surface of the parent tube.

c. Any insights on why the circumferential signals in the three tubes installed during 2R20 (2018) were not observed during the inspections performed during 2R21 (2020). The staff notes that the Ghent Version 2 probe was also used during 2R21 and that two sleeved tubes were plugged due to axial scratches located in the lower tubesheet sleeve joint (ML20287A373).

d. It is the staff's understanding that the use of a high-speed buffing tool for mechanically conditioning the tube prior to sleeving may have been eliminated prior to 2R20 (2018) (ML18348B206). If so, could the lack of tube buffing prior to sleeve installation in these tubes be related to the eddy current indications? In addition, are any changes to the sleeve installation process being considered?