Energy Harbor Nuclear Corp.
Beaver Valley Power Station
P. O. Box 4
Shippingport, PA 15077

Rod L. Penfield
Site Vice President, Beaver Valley Nuclear
724-682-5234

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L-20-117

10 CFR 50.55a

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
10 CFR 50.55a Request Number: 2-TYP-4-RV-06, Hardship for Hot Leg Nozzle Inspections

In accordance with 10 CFR 50.55a(z)(2), Energy Harbor Nuclear Corp. hereby requests Nuclear Regulatory Commission (NRC) approval of request 2-TYP-4-RV-06 that proposes to eliminate the examination of the Beaver Valley Power Station, Unit No. 2 (BVPS-2) reactor vessel hot leg nozzle-to-safe end dissimilar metal welds 2RCS-REV21-N-24, 2RCS-REV21-N-26, and 2RCS-REV21-N-28.

As a result of the hardship produced by the recent pandemic and the resulting national state of emergency, Energy Harbor Nuclear Corp. is requesting expedited approval of 2-TYP-4-RV-06. The proposed alternative will eliminate performance of the examination during 2R21 and resume the normal outage examination frequency at the next opportunity, currently expected to be the next refueling outage (2R22) set to begin on October 10, 2021.

To support the startup and critical generation of BVPS-2 from its scheduled refuel outage, Energy Harbor Nuclear Corp. requests approval of the proposed alternative by April 12, 2020.

The enclosed request identifies the affected components, applicable code requirements, and a description and basis for the proposed alternative.

There are no regulatory commitments contained in this submittal. If there are any questions or additional information is required, please contact Mr. Phil H. Lashley, Acting Manager – Nuclear Licensing and Regulatory Affairs, at (330) 315-6808.

Sincerely,

Rod L. Penfield

Enclosure:

Beaver Valley Power Station, Unit No. 2,
10 CFR 50.55a Request Number: 2-TYP-4-RV-06

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

--Hardship or Unusual Difficulty
without Compensating Increase in Level of Quality or Safety--

1. ASME Code Component(s) Affected

Component Numbers:	Beaver Valley Power Station, Unit No. 2 (BVPS-2) reactor vessel (RV) hot leg nozzle-to-safe end dissimilar metal (DM) welds (2RCS-REV21-N-24, 2RCS-REV21-N-26, and 2RCS-REV21-N-28)
Code Class:	Class 1
Examination Category:	Class 1 PWR Pressure Retaining Dissimilar Metal Piping and Vessel Nozzle Butt Welds Containing Alloy 82/182 (ASME Code Case N-770-2, Table 1) and Class 1 PWR Components Containing Alloy 600/82/182 (ASME Code Case N-722-1, Table 1)
Item Number:	A-2 in Code Case N-770-2 and B15.90 in Code Case N-722-1
Description:	Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds (Code Case N-770-2) and Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials (Code Case N-722-1)

2. Applicable Code Edition and Addenda

American Society of Mechanical Engineers (ASME), Boiler Pressure Vessel (BPV) Code, Division 1, Section XI, 2013 Edition.

ASME BPV Code Case N-770, Revision 2, “Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Metal With or Without Application of Listed Mitigation Activities.”

ASME BPV Code Case N-722, Revision 1, "Additional Examinations of PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials."

3. Applicable Code Requirement

10 CFR 50.55a(g)(6)(ii)(F) states the following: "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–770–2 instead of ASME BPV Code Case N–770–1, subject to the conditions specified in paragraphs (g)(6)(ii)(F)(*2*) through (*13*) of this section…"

ASME Code Case N-770-2 states in Table 1, "Examination Categories," for item A-2 "Unmitigated butt weld at Hot Leg operating Temperature ≤ 625°F," that the weld surface must be visually examined each refueling outage.

10 CFR 50.55a(g)(6)(ii)(E) states the following: "All licensees of pressurized water reactors must augment their inservice inspection program by implementing ASME Code Case N–722–1, subject to the conditions specified in paragraphs (g)(6)(ii)(E)(*2*) through (*4*) of this section."

ASME Code Case N-722-1 states in Table 1, "Examination Categories," for item B15.90 "Hot leg nozzle-to-pipe connections," that the weld surface must be visually examined each refueling outage.

4. Reason for Request

Beaver Valley Power Station Unit 2 (BVPS-2) is scheduled to start its 21st refueling outage (2R21) on April 12, 2020. The hot leg nozzle visual examination is required to be performed in 2R21, as specified by ASME Code Case N-770-2 and ASME Code Case N-722-1. For the reasons specified below, performance of the hot leg nozzle visual examinations creates a hardship due to expected challenges with obtaining and maintaining staffing levels sufficient to perform the examinations during 2R21. Elimination of these examinations could reduce the risk of exposure for critical contract and direct hire personnel to the COVID-19 virus.

On March 13, 2020, the President of the United States declared a national emergency due to the spread and infectious nature of the Coronavirus-2019 (COVID-19) virus and resulting pandemic. The most recent guidance from the Centers for Disease Control and Prevention (CDC) includes recommendations for social distancing by maintaining approximately six feet from other personnel to limit the spread of the virus. On March 28, 2020, the Governor of Pennsylvania issued a Stay at Home order for Beaver County and the surrounding counties of Allegheny and Butler. Furthermore, on March 28,

2020, the Department of Homeland Security identified workers in the nuclear energy sector as essential critical infrastructure workers.

To prevent the spread of COVID-19 at Beaver Valley Power Station (BVPS), and to protect the health and safety of plant personnel while maintaining responsibilities to support critical infrastructure, Energy Harbor Nuclear Corp. intends to reduce the amount of personnel on-site, which will pose a hardship for completing the currently planned 2R21 refueling outage work scope. Energy Harbor Nuclear Corp. is also contingency planning in case some of its workforce becomes unavailable due to the COVID-19 outbreak. With the current work scope and potential loss of personnel, the company may not be able to complete the refueling outage in a timely manner, which could negatively impact critical infrastructure that is needed during this time.

This request is submitted due to the expected hardship of obtaining and maintaining onsite staff sufficient to prepare, perform, and recover from this examination. At BVPS-2, this exam requires construction trades to open hatches in the floor of the refueling cavity, install temporary lighting, remove neutron shield material, and remove insulation. Additional contract and onsite staff are required to perform radiological surveys and the weld examinations. Because of the rapid spread and infection rates of the virus, BVPS anticipates challenges to maintain staff levels throughout the outage and is requesting relief where appropriate to reduce necessary staff and ensure being able to return the unit to power production in a reasonable time to support the power needs of the surrounding area.

5. Proposed Alternative and Basis for Use

BVPS-2 is requesting relief to not perform the reactor vessel hot leg nozzle visual examinations during the 2R21 refueling outage. The visual examinations of the reactor vessel hot leg nozzles will be performed during the next refueling outage (2R22) and subsequent refueling outages in accordance with the requirements of ASME Code Cases N-770-2 and N-722-1.

The proposed alternative is based on past BVPS-2 inspection results, a review of industry operating experience related to hot leg nozzle primary water stress corrosion cracking (PWSCC) indications, compensatory actions that would detect leakage if it were to occur, and the chemical mitigation benefits that result from zinc addition to the RCS.

The primary degradation mechanism addressed by the examinations of ASME Code Cases N-770-2 and N-722-1 is PWSCC. This degradation mechanism occurs when a susceptible material is exposed to a primary water environment, elevated stress levels, and high operating temperatures.

Hot leg nozzle examinations were performed at BVPS-2 during the 2R14 refueling outage (fall 2009), 2R15 refueling outage (spring 2011), 2R16 refueling outage (fall 2012), 2R18 refueling outage (fall 2015), and 2R19 refueling outage (spring 2017). No evidence of pressure boundary leakage or corrosion of ferritic steel components was identified during any of the visual and ultrasonic examinations.

An assessment of recent industry operating experience (OE) revealed that with respect to reactor vessel hot leg nozzle weld examinations, PWSCC flaws are generally discovered using ultrasonic examination techniques. During the last BVPS-2 refueling outage (2R20 in fall 2018), full coverage was obtained with both volumetric (ultrasonic) and eddy current examinations of the reactor vessel hot leg nozzle welds. The examinations were performed in accordance with the requirements of ASME Code Case N-770-2, with no flaws identified. The industry hot leg nozzle PWSCC indications that have been discovered to date were found prior to the issuance of ASME Code Case N-770-2. Since the ultrasonic examinations at BVPS-2 are performed at the frequency required by ASME code case N-770-2, the relevant ultrasonic examination OE has been adequately addressed to ensure that any PWSCC flaws in the BVPS-2 reactor vessel hot leg nozzles would be discovered and mitigated before they would become a safety concern.

A review of recent industry OE also revealed that in one case a visual examination technique did identify leakage in a reactor vessel nozzle weld. However, this was the first flaw identified in a reactor vessel hot leg nozzle weld and it was identified prior to the time that more routine examinations were required. Additionally, in the one case where a visual examination of a reactor vessel hot leg nozzle identified a through-wall flaw, there was significant weld repairs performed in the nozzle weld during fabrication that would have likely increased the PWSCC susceptibility of the hot leg nozzle weld. A review of immediately available records for BVPS-2 did not identify any known weld repairs within the BVPS-2 reactor vessel hot leg nozzle welds.

Zinc addition to the primary reactor coolant was implemented at BVPS-2 in October of 2010, and the zinc deposits have been building since that time. Zinc addition decreases the PWSCC susceptibility of alloy 600/82/182 components in the reactor coolant system, including the reactor vessel hot leg nozzles. The progress of the zinc buildup on the primary system surfaces is measured in terms of ppb-months, which is the product of the concentration of zinc in the chemical mitigation, and the time over which it has been applied. In October 2017, the plant reached 300 ppb-months, at which time significant chemical mitigation against PWSCC was achieved. This decreases the PWSCC susceptibility of the BVPS-2 reactor vessel hot leg nozzles.

Without inspecting the hot leg nozzle welds directly, there are other inspections and equipment in the area that would identify a leak should one occur. The general area around and below the reactor is examined as part of the pressure test program walkdown during mode 3 start up. Any leakage noted would be investigated to

determine the source. During operation, an increase in radiation levels within containment would be noted if there were significant leakage.

Also, BVPS-2 has an integrated leakage monitoring program that monitors RCS leakage. The Unit 2 reactor coolant system (RCS) inventory is calculated by procedures 2OST-6.2 or 2OST-6.2A reactor coolant system water inventory balance, with a surveillance test requirement to be performed every 72 hours. The RCS Integrated Leakage Program (1/2-ADM-0710) provides guidance where the RCS leakage is quantified and compared to the recent history of RCS leakage to better understand if current changes are outside normally expected values.

The RCS Integrated Leakage Program includes guidance to determine if any program action level criteria are exceeded. The action level criteria are provided in the following table.

Action Level	***Action Level Criteria***
Level 1	*Is the rolling average of the last seven performances of Unidentified RCS leak rate greater than 0.1 gpm?*
Level 1	*Are the last nine consecutive Unidentified RCS leak rates greater than baseline mean?*
Level 2	*Are the last two consecutive Unidentified RCS leak rates greater than 0.15 gpm?*
Level 2	*Are two of the last three consecutive Unidentified RCS leak rates greater than [mean Unidentified RCS Leakage + 2 Standard Deviation]?*
Level 3	*Is this Unidentified RCS leak rate greater than 0.3 gpm?*
Level 3	*Is this Unidentified RCS leak rate greater than [mean + 3 Standard Deviation]?*

The 0.1 gpm is consistent with WCAP-16465-NP, Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors and is one-tenth of the technical specification (TS) limit for unidentified leakage. The RCS Integrated Leakage Program includes the requirement to identify the leakage source and could include entering containment to identify the source of the leakage. If the source of the leakage is found and isolated, the program directs operation personnel to re-perform an RCS leak rate calculation to confirm that the source of leakage has been addressed.

The RCS leakage quantity is reviewed against the TS associated with RCS leakage criteria. Depending on the source identified, a shutdown could be required in

accordance with TS Limiting Condition for Operation (LCO) 3.4.13 that has the following specific limits:

a. No pressure boundary LEAKAGE;
b. 1 gpm unidentified LEAKAGE;
c. 10 gpm identified LEAKAGE; and
d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

A through-wall leak from the reactor vessel hot leg nozzle weld would constitute pressure boundary leakage.

Should any of these limitations be exceeded, the appropriate LCO Condition would be entered and the required actions performed within the specified completion time, including plant shutdown if required.

Based on previous BVPS-2 examination results, a review of relevant industry operating experience, compensatory actions that would detect leakage if it were to occur, the benefits of zinc addition, and the ability to detect leakage while the plant is operating, it is determined that the structural integrity of the BVPS-2 reactor vessel hot leg nozzle welds will be maintained even if the visual exams are not performed during the 2R21 refueling outage.

6. Duration of Proposed Alternative

BVPS-2 is requesting relief to eliminate performance of this examination during 2R21 and resume the normal outage examination frequency at the next opportunity, currently expected to be the next refueling outage (2R22) set to begin fall 2021.

7. References

1. Code Case N-770-2, “Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material with or Without Application of listed Mitigation Activities,” Section XI, Division 1.

2. Code Case N-722-1, “Additional Examinations of PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials,” Section XI, Division 1.