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

May 2, 2022

Mr. John J. Grabnar
Site Vice President
Energy Harbor Nuclear Corp.
Beaver Valley Power Station
Mail Stop P-BV-SSEB
P.O. Box 4, Route 168
Shippingport, PA 15077-0004

SUBJECT: BEAVER VALLEY POWER STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 314 AND 204 RE: REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-577, REVISION 1, "REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS" (EPID L-2021-LLA-0220)

Dear Mr. Grabnar:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 314 to Renewed Facility Operating License No. DPR-66 for the Beaver Valley Power Station (Beaver Valley), Unit 1, and Amendment No. 204 to Renewed Facility Operating License No. NPF-73 for the Beaver Valley, Unit 2. These amendments consist of changes to the technical specifications in response to your application dated September 15, 2021, as supplemented by letter dated February 4, 2022.

The amendments adopt Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections," which is an approved change to the Standard Technical Specifications, into the Beaver Valley, Unit Nos. 1 and 2, technical specifications.

A copy of the related safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

/RA/

Brent Ballard, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment No. 314 to DPR-66
2. Amendment No. 204 to NPF-73
3. Safety Evaluation

cc: Listserv

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

ENERGY HARBOR NUCLEAR CORP.

ENERGY HARBOR NUCLEAR GENERATION LLC

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 314
Renewed License No. DPR-66

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Energy Harbor Nuclear Corp., acting on its own behalf and as agent for Energy Harbor Nuclear Generation LLC* (the licensees), dated September 15, 2021, as supplemented by letter dated February 4, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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* Energy Harbor Nuclear Corp. is authorized to act as agent for Energy Harbor Nuclear Generation LLC and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-66 is hereby amended to read as follows:

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 314, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: May 2, 2022

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

ENERGY HARBOR NUCLEAR CORP.

ENERGY HARBOR NUCLEAR GENERATION LLC

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 204
Renewed License No. NPF-73

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Energy Harbor Nuclear Corp., acting on its own behalf and as agent for Energy Harbor Nuclear Generation LLC* (the licensees), dated September 15, 2021, as supplemented by letter dated February 4, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I.

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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* Energy Harbor Nuclear Corp. is authorized to act as agent for Energy Harbor Nuclear Generation LLC and has exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-73 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 204, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto and hereby incorporated in the license. Energy Harbor Nuclear Corp. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

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Date: 2022.05.02 12:12:18 -04'00'

James G. Danna, Chief
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: May 2, 2022

ATTACHMENT TO LICENSE AMENDMENT NOS. 314 AND 204

BEAVER VALLEY POWER STATION, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-66 AND NPF-73

DOCKET NO. 50-334 and 50-412

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Renewed Facility Operating License No. DPR-66

Remove	Insert
Page 3	Page 3

Renewed Facility Operating License No. NPF-73

Remove	Insert
Page 4	Page 4

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Appendix A, Technical Specifications

Remove	Insert
5.5 - 4	5.5 - 4
5.5 - 5	5.5 - 5
5.5 - 6	5.5 - 6
5.5 - 7	5.5 - 7
5.5 - 8	5.5 - 8
5.5 - 9	5.5 - 9
5.5 - 11	5.5 - 11
5.5 - 12	5.5 - 12
5.6 - 4	5.6 - 4
5.6 - 4a	
5.6 - 5	5.6 - 5
5.6 - 6	5.6 - 6

(3) Energy Harbor Nuclear Corp., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Energy Harbor Nuclear Corp., pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Energy Harbor Nuclear Corp., pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Energy Harbor Nuclear Corp. is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 314, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Auxiliary River Water System

(Deleted by Amendment No. 8)

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations set forth in 10 CFR Chapter 1 and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Energy Harbor Nuclear Corp. is authorized to operate the facility at a steady state reactor core power level of 2900 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 204, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. Energy Harbor Nuclear Corp. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

5.5 Programs and Manuals

5.5.5 Steam Generator (SG) Program

An SG Program for Unit 1 and Unit 2 shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the SG Program for Unit 1 shall include the provisions of Specification 5.5.5.1 and the SG Program for Unit 2 shall include the provisions of Specification 5.5.5.2.

5.5.5.1 Unit 1 SG Program

a. Provisions for Condition Monitoring Assessments

Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. Provisions for Performance Criteria for SG Tube Integrity

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

5.5 Programs and Manuals

5.5.5.1 Unit 1 SG Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is also not to exceed 1 gpm per SG, except during a SG tube rupture.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG Tube Plugging Criteria

Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG Tube Inspections

Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 96 effective full power months, which defines the inspection period.

5.5 Programs and Manuals

5.5.5.1 Unit 1 SG Program (continued)

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program

a. Provisions for Condition Monitoring Assessments

Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Provisions for Performance Criteria for SG Tube Integrity

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and, except for flaws addressed through application of the alternate repair criteria discussed in Specification 5.5.5.2.c.4, a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   When alternate repair criteria discussed in Specification 5.5.5.2.c.4 are applied to axially oriented outside diameter stress corrosion cracking indications at tube support plate locations, the probability that one or more of these indications in a SG will burst under postulated main steam line break conditions shall be less than 1×10^{-2}.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

SGs and leakage rate for an individual SG. Except during a SG tube rupture, leakage from all sources excluding the leakage attributed to the degradation described in Specification 5.5.5.2.c.4 is also not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, “RCS Operational LEAKAGE.”

c. Provisions for SG Tube Plugging or Repair Criteria

1. Tubes found by inservice inspection to contain a flaw in a non-sleeved region with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate plugging or repair criteria discussed in Specification 5.5.5.2.c.4 or 5.5.5.2.c.5.

2. Tubes found by inservice inspection to contain a flaw in a sleeve (excluding the sleeve to tube joint) with a depth equal to or exceeding the following percentages of the nominal sleeve wall thickness shall be plugged:

ABB Combustion Engineering TIG welded sleeves	27%
Westinghouse laser welded sleeves	25%
Westinghouse leak limiting Alloy 800 sleeves	Any flaw

3. Tubes with a flaw in a sleeve to tube joint shall be plugged.

4. Tube support plate voltage-based plugging or repair criteria may be applied as an alternative to the 40% depth based criteria of Specification 5.5.5.2.c.1.

Tube Support Plate Plugging Limit is used for the disposition of an Alloy 600 SG tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging or repair limit is described below:

a) SG tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

b) SG tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be plugged or repaired, except as noted in 5.5.5.2.c.4.c below.

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

c) SG tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit (calculated according to the methodology in Generic Letter 95-05 as supplemented) may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation.

d) SG tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the upper voltage repair limit (calculated according to the methodology in Generic Letter 95-05 as supplemented) will be plugged or repaired.

e) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits specified in 5.5.5.2.c.4.a through 5.5.5.2.c.4.d.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled SG inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 24 effective full power months, which defines the inspection period.

3. Indications left in service as a result of application of the tube support plate voltage-based plugging or repair criteria (Specification 5.5.5.2.c.4) shall be inspected by bobbin coil probe during all future refueling outages.

   Implementation of the SG tube-to-tube support plate plugging or repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

4. When the F* methodology has been implemented, inspect 100% of the inservice tubes in the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube plugging or repair criteria of Specification 5.5.5.2.c.5 every 24 effective full power months or one interval between refueling outages (whichever is less).

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

5. For Alloy 800 sleeves: The parent tube, in the area where the sleeve-to-tube hard roll joint and the sleeve-to-tube hydraulic expansion joint will be established, shall be inspected prior to installation of the sleeve. Sleeve installation may proceed only if the inspection finds these regions free from service induced indications.

e. Provisions for monitoring operational primary to secondary LEAKAGE

f. Provisions for SG Tube Repair Methods

SG tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

1. ABB Combustion Engineering TIG welded sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.

2. Westinghouse laser welded sleeves, WCAP-13483, Revision 2.

3. Westinghouse leak-limiting Alloy 800 sleeves, WCAP-15919-P, Revision 2.

5.6 Reporting Requirements

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

WCAP-18124-NPA, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2012, may be used as an alternative to Section 2.2 of WCAP-14040-A, Revision 4.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.5 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.6 Steam Generator (SG) Tube Inspection Report

5.6.6.1 Unit 1 SG Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.1, "Unit 1 SG Program." The report shall include:

a. The scope of inspections performed on each SG;

b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

c. For each degradation mechanism found:

   1. The nondestructive examination techniques utilized;

   2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

   3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment; and

   4. The number of tubes plugged during the inspection outage.

d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

5.6 Reporting Requirements

5.6.6. Steam Generator (SG) Tube Inspection Report (continued)

5.6.6.1 Unit 1 SG Tube Inspection Report (continued)

e. The number and percentage of tubes plugged to date, and the effective plugging percentage in each SG; and

f. The results of any SG secondary side inspections.

5.6.6.2 Unit 2 SG Tube Inspection Report

1. A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.2, "Unit 2 SG Program." The report shall include:

   a. The scope of inspections performed on each SG;

   b. The nondestructive examination techniques utilized for tubes with increased degradation susceptibility;

   c. For each degradation mechanism found:

      1. The nondestructive examination techniques utilized;

      2. The location, orientation (if linear), measured size (if available), and voltage response for each indication. For tube wear at support structures less than 20 percent through-wall, only the total number of indications needs to be reported;

      3. A description of the condition monitoring assessment and results, including the margin to the tube integrity performance criteria and comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment;

      4. The number of tubes plugged or repaired during the inspection outage; and

      5. The repair methods utilized and the number of tubes repaired by each repair method.

   d. An analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection (the forward-looking tube integrity assessment) relative to the applicable performance criteria, including the analysis methodology, inputs, and results;

5.6 Reporting Requirements

5.6.6 Steam Generator (SG) Tube Inspection Report (continued)

5.6.6.2 Unit 2 SG Tube Inspection Report (continued)

e. The number and percentage of tubes plugged or repaired to date, and the effective plugging percentage in each SG; and

f. The results of any SG secondary side inspections.

2. A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.2, “Unit 2 SG Program,” when voltage-based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

3. For implementation of the voltage-based plugging or repair criteria to tube support plate intersections, notify the Commission prior to returning the SG to service (MODE 4) should any of the following conditions arise:

   a. If circumferential crack-like indications are detected at the tube support plate intersections.

   b. If indications are identified that extend beyond the confines of the tube support plate.

   c. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

4. A report shall be submitted within 90 days after the initial entry into MODE 4 following an outage in which the F* methodology was applied. As applicable, the report shall include the following hot-leg and cold-leg tubesheet region inspection results associated with the application of F*:

   a. Total number of indications, location of each indication, orientation of each indication, severity of each indication, and whether the indications initiated from the inside or outside surface.

   b. The cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet.

   c. The projected end-of-cycle accident-induced leakage from tubesheet indications.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 314 AND 204 TO RENEWED

FACILITY OPERATING LICENSES NOS. DPR-66 AND NPF-73

ENERGY HARBOR NUCLEAR CORP.

ENERGY HARBOR NUCLEAR GENERATION LLC

BEAVER VALLEY POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By application dated September 15, 2021, as supplemented by letter dated February 4, 2022, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML21258A319 and ML22035A122, respectively), Energy Harbor Nuclear Corp. (the licensee) requested changes to the technical specifications (TSs) for Beaver Valley Power Station (Beaver Valley), Units 1 and 2. The licensee requested that the U.S. Nuclear Regulatory Commission (NRC, the Commission) process the proposed amendment under the Consolidated Line Item Improvement Process (CLIIP). The proposed changes would revise the "Steam Generator (SG) Program" and the "Steam Generator Tube Inspection Report" TSs based on Technical Specifications Task Force (TSTF) Traveler TSTF-577, Revision 1, "Revised Frequencies for Steam Generator Tube Inspections" (TSTF-577) (ADAMS Accession No. ML21060B434), and the associated NRC staff safety evaluation (SE) of TSTF-577 (ADAMS Accession No. ML21098A188).

Throughout the application, the licensee refers to TSTF-577-A as a basis for the requested amendment. The NRC staff notes that the "-A" designation added to TSTF-577 is an industry convention used to indicate that the traveler has been approved by the NRC. TSTF-577 and TSTF-577-A are the same document. However, since TSTF-577-A is not an NRC designation, this SE refers to the technical specification change traveler as TSTF-577.

The tubes within an SG function as an integral part of the reactor coolant pressure boundary and, in addition, isolate fission products in the primary coolant from the secondary coolant and the environment. SG tube integrity means that the tubes are capable of performing this safety function in accordance with the plant design and licensing basis.

Beaver Valley has two units. The Unit 1 SGs have Alloy 690 thermally treated (Alloy 690TT) tubes. The Unit 2 SGs have Alloy 600 mil-annealed (Alloy 600MA) tubes.

The supplemental letter dated February 4, 2022, provided additional information that clarified the application, did not expand the scope of the application as noticed, and did not change the NRC staff's proposed no significant hazards consideration determination as published in the *Federal Register* on February 22, 2022 (87 FR 9651).

1.1 Proposed TS Changes to Adopt TSTF-577

In accordance with NRC staff-approved TSTF-577, the licensee proposed changes that would revise Beaver Valley TS 5.5.5, "Steam Generator (SG) Program," and TS 5.6.6, "Steam Generator (SG) Tube Inspection Report." Specifically, the licensee proposed the following changes to adopt TSTF-577:

TS 5.5.5, "Steam Generator (SG) Program":

- The introductory paragraph to TS 5.5.5 would be revised by replacing "Steam Generator" with "SG" in several instances.
- TS 5.5.5.1, "Unit 1 SG Program," and TS 5.5.5.2, "Unit 2 SG Program," paragraph b.1, would be revised by replacing "steam generator" with "SG" in several instances.
- TS 5.5.5.1.d.2 and TS 5.5.5.2.d.2 would be revised by deleting the requirement to base inspection frequency on the more restrictive metric between either the effective full power months (EFPM) or refueling outage and to use just the EFPM metric.
- TS 5.5.5.1.d.2 and TS 5.5.5.2.d.2 would be revised by deleting the allowance to extend the inspection period by 3 EFPM and by deleting the discussion of prorating inspections.
- TS 5.5.5.1.d.2 would be revised by deleting the requirement to inspect 100 percent of the tubes during each period in paragraphs d.2.a, d.2.b, d.2.c, and d.2.d (144, 120, 96, and 72 EFPM, respectively) and by adding the requirement to inspect 100 percent of the tubes every 96 EFPM.
- TS 5.5.5.2.d.2 would be revised by changing the requirement to inspect 100 percent of the tubes from every 60 EFPM to every 24 EFPM.
- TS 5.5.5.1.d.3 would be revised by replacing "shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections)" with "shall be at the next refueling outage."

TS 5.6.6, "Steam Generator Tube Inspection Report" (note that changes described below apply to both TS 5.6.6.1, "Unit 1 SG Tube Inspection Report," and TS 5.6.6.2, "Unit 2 SG Tube Inspection Report," except as noted):

- Existing reporting requirement b. would be renumbered as c. and be revised by editorial and punctuation changes.
- New reporting requirement b. would be added to require the nondestructive examination (NDE) techniques utilized for tubes with increased degradation susceptibility be reported.

- Existing reporting requirement c. would be renumbered as c.1. and be revised by editorial and punctuation changes.

- Existing reporting requirement d. would be renumbered as c.2. and be revised to note that the location, orientation (if linear), measured size (if available), and voltage response do not need to be reported for tube wear indications at support structures that are less than 20 percent through-wall. However, the total number of tube wear indications at support structures that are less than 20 percent through-wall would be reported.

- New reporting requirement d. would be added to require an analysis summary of the tube integrity conditions predicted to exist at the next scheduled inspection relative to the applicable performance criteria, including the analysis, methodology, inputs, and results.

- Existing reporting requirement e. would be renumbered as c.4. and be revised by editorial and punctuation changes.

- Existing reporting requirement f. would be renumbered as e. and be revised by editorial and punctuation changes.

- New reporting requirement f. would be added to require the results of any SG secondary side inspections be reported.

- Existing reporting requirement g. would be renumbered as c.3. and be revised to add the requirements to report a description of the condition monitoring assessment, the margin to the tube integrity performance criteria, and a comparison with the margin predicted to exist at the inspection by the previous forward-looking tube integrity assessment. In addition, the requirement to report the results of tube pulls and in-situ testing would be deleted.

- Existing reporting requirement h. would be renumbered as c.5. and be revised by editorial changes (only applies to Unit 2).

1.2 Additional Proposed TS Changes

In addition to the changes proposed consistent with TSTF-577, Revision 1 discussed in Section 1.1, the licensee proposed the following variations.

1.2.1 Editorial Variations

The licensee identified two variations. For the first variation, the licensee replaced the words “Steam Generator” with the letters “SG” in Beaver Valley TS 5.5.5, TS 5.5.5.1, TS 5.5.5.2, TS 5.6.6.1, and TS 5.6.6.2. For the second variation, the licensee noted that Beaver Valley TSs have different numbering than standard technical specifications (STSs) on which TSTF-577 was based. Specifically, the “Steam Generator (SG) Program” is numbered 5.5.5/5.5.5.1 in Beaver Valley Unit 1 and 5.5.5/5.5.5.2 in Beaver Valley Unit 2 TSs rather than 5.5.9 as stated in TSTF-577. Additionally, the “Steam Generator Tube Inspection Report” is numbered 5.6.6/5.6.6.1 for Beaver Valley Unit 1 and 5.6.6/5.6.6.2 in Beaver Valley Unit 2 TSs rather than 5.6.7 as stated in TSTF-577.

1.2.2 Other Variations

The licensee noted that Beaver Valley Unit 1 TS 5.5.5.1, paragraph d.2 contains a one-time deferral of SG tube inspections from the spring of 2021 refueling outage (1R27) to the fall of 2022 refueling outage (1R28). The licensee proposed to delete the one-time deferral.

2.0 REGULATORY EVALUATION

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.36(c)(5), "Administrative controls," state that "[a]dministrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in [10 CFR] 50.4." Technical Specification Section 5.0, "Administrative Controls," requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TS to operate the facility in a safe manner.

The NRC staff's guidance for the review of TSs is in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition" (SRP), Chapter 16.0, "Technical Specifications," Revision 3, dated March 2010 (ADAMS Accession No. ML100351425). As described therein, as part of the regulatory standardization effort, the NRC staff has prepared STSs for each of the LWR nuclear designs. Accordingly, the NRC staff's review includes consideration of whether the proposed changes are consistent with NUREG-1431[1], as modified by NRC-approved travelers.

TSTF-577 revised the STSs related to SG tube inspections and SG tube inspection reporting requirements. The NRC approved TSTF-577, under the CLIIP on April 14, 2021 (ADAMS Package Accession No. ML21099A086).

3.0 TECHNICAL EVALUATION

3.1 Proposed TS Changes to Adopt TSTF-577

The NRC staff compared the licensee's proposed TS changes in Section 1.1 of this SE against the changes approved in TSTF-577. In accordance with SRP Chapter 16.0, the NRC staff determined that the STS changes approved in TSTF-577 are applicable because Beaver Valley is a pressurized-water reactor design plant and the NRC staff approved the TSTF-577 changes for pressurized-water reactor designs. The NRC staff finds that the licensee's proposed changes to the Beaver Valley TSs in Section 1.1 of this SE are consistent with those found acceptable in TSTF-577.

In the SE of TSTF-577, the NRC staff concluded that the TSTF-577 changes to STS 5.5.9, "Steam Generator (SG) Program," and STS 5.6.7, "Steam Generator Tube Inspection Report," were acceptable because, as discussed in Section 3.0 of that SE, they continued to ensure SG tube integrity and, therefore, protected the public health and safety. In particular, the structural

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[1] U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431, Volume 1, "Specifications," and Volume 2, "Bases," Revision 5, September 2021 (ADAMS Accession Nos. ML21259A155 and ML21259A159, respectively).

integrity performance criterion and accident-induced leakage performance criterion (explained in STS 5.5.9.b, items 1 and 2, respectively) will continue to be met with the proposed revised SG inspection intervals (maximum allowable time between SG inspections) and inspection periods (maximum allowable time between 100 percent of SG tubes inspections). Additionally, the proposed changes to the reporting requirements will provide more detailed and consistent information to the NRC. Therefore, the NRC staff found that the proposed changes to the SG program and inspection reporting requirements were acceptable because they continued to meet the requirements of 10 CFR 50.36(c)(5) by providing administrative controls necessary to assure operation of the facility in a safe manner. For these same reasons, the NRC staff concludes that the corresponding proposed changes to the Beaver Valley TSs in Section 1.1 of this SE continue to meet the requirements of 10 CFR 50.36(c)(5).

3.2 Additional Proposed TS Changes

3.2.1 Editorial Variations

The licensee identified two variations. For the first variation, the licensee replaced the words “Steam Generator” with the letters “SG” in Beaver Valley TS 5.5.5, TS 5.5.5.1, TS 5.5.5.2, TS 5.6.6.1, and TS 5.6.6.2. For the second variation, the licensee noted that Beaver Valley TS has different numbering than STS. Specifically, the “Steam Generator (SG) Program” is numbered 5.5.5/5.5.5.1 in Beaver Valley Unit 1 and 5.5.5/5.5.5.2 in Beaver Valley Unit 2 TSs rather than 5.5.9 as stated in TSTF-577. Additionally, the “Steam Generator Tube Inspection Report” is numbered 5.6.6/5.6.6.1 for Beaver Valley Unit 1 and 5.6.6/5.6.6.2 in Beaver Valley Unit 2 TSs rather than 5.6.7 as stated in TSTF-577. The NRC staff finds the use of “SG” to replace “Steam Generator” and the use of different TS numbering acceptable because they do not substantively alter TS requirements.

3.2.2 Other Variation

The licensee noted that Beaver Valley Unit 1 TS 5.5.5.1, paragraph d.2 contains a one-time deferral of SG tube inspections from the spring of 2021 refueling outage (1R27) to the fall of 2022 refueling outage (1R28). The licensee proposed to delete the one-time deferral because upon implementation of TSTF-577 prior to refueling outage 1R28, the one-time deferral is no longer needed. Therefore, because the one-time deferral is no longer needed, the NRC staff finds that deletion of the Beaver Valley Unit 1 TS 5.5.5.1, paragraph d.2 one-time deferral is acceptable.

3.3 TS Change Consistency

The NRC staff reviewed the proposed TS changes for technical clarity and consistency with the existing requirements for customary terminology and formatting. The NRC staff finds that the proposed changes are consistent with Chapter 16.0 of the SRP and are therefore acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments on March 25, 2022. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment relates, in part, to changes in recordkeeping, reporting, or administrative procedures or requirements. The amendment also relates, in part, to changing requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on February 22, 2022 (87 FR 9651). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Clinton Ashley, NRR

Date: May 2, 2022

SUBJECT: BEAVER VALLEY POWER STATION, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 314 AND 204 RE: REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-577, REVISION 1, "REVISED FREQUENCIES FOR STEAM GENERATOR TUBE INSPECTIONS" (EPID L-2021-LLA-0220) DATED MAY 2, 2022

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