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

March 22, 2021

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

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT NO. 310 RE: ONE-TIME EXTENSION OF UNIT NO. 1 STEAM GENERATOR INSPECTIONS [COVID-19] (EPID L-2021-LLL-0002)

Dear Mr. Grabnar:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 310 to Renewed Facility Operating License No. DPR 66 for the Beaver Valley Power Station, Unit No. 1. This amendment consists of changes to the technical specifications (TSs) in response to your application dated January 27, 2021.

The amendment revises TS 5.5.5.1, “Unit 1 SG [Steam Generator] Program,” paragraph d.2 to defer the spring 2021 refueling outage (1R27) SG inspections to the fall 2022 refueling outage (1R28).

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/***

Jennifer C. Tobin, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-334

Enclosures:
1. Amendment No. 310 to DPR-66
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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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. 310
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 January 27, 2021, 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. 310, 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 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James G. Danna

Digitally signed by James G. Danna
Date: 2021.03.22 09:04:06 -04'00'

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

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

Date of Issuance: March 22, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 310

BEAVER VALLEY POWER STATION, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Renewed Facility Operating License No. DPR-66

| Remove | Insert |
|---|---|
| Page 3 | Page 3 |

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

Appendix A, Technical Specifications

| Remove | Insert |
|---|---|
| 5.5 - 5 | 5.5 - 5 |

(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. 310, 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)

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 each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections).* In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type

* The spring of 2021 (1R27) refueling outage SG inspections may be deferred to the fall of 2022 (1R28) refueling outage.

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

# SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

# RELATED TO AMENDMENT NO. 310 TO RENEWED

# FACILITY OPERATING LICENSE NO. DPR-66

# ENERGY HARBOR NUCLEAR CORP.

# ENERGY HARBOR NUCLEAR GENERATION LLC

# BEAVER VALLEY POWER STATION, UNIT NO. 1

# DOCKET NO. 50-334

## 1.0 INTRODUCTION

By application dated January 27, 2021 (Reference 1), Energy Harbor Nuclear Corp. (Energy Harbor, the licensee)[1] submitted a license amendment request to the U.S. Nuclear Regulatory Commission (NRC, the Commission), requesting a one-time deferral of steam generator (SG) inspections for the Beaver Valley Power Station, Unit No. 1 (Beaver Valley 1) spring 2021 outage (1R27).

The spring 2021 refueling outage is scheduled to begin in April and would require approximately 140 people onsite working in close proximity for extended periods of time to perform SG inspections. On January 31, 2020, the U.S. Department of Health and Human Services declared a public health emergency for the United States to aid the nation's healthcare community in responding to the Coronavirus Disease 2019 (COVID-19). Subsequently, the COVID-19 outbreak was characterized as a pandemic by the World Health Organization on March 11, 2020, and, on March 13, 2020, the President of the United States of America declared the COVID-19 pandemic a national emergency.

The license amendment request proposed the addition of a footnote in Technical Specification (TS) 5.5.5.1, "Unit 1 SG Program," paragraph d.2 to allow a one-time deferral of SG inspections to the fall 2022 refueling outage (1R28). The stated purpose for the proposed license amendment request is to protect personnel safety and to preclude the potential for transmittal and spread of COVID-19.

The NRC staff's proposed no significant hazards consideration determination was published in the *Federal Register* on February 11, 2021 (86 FR 9087). The final no significant hazards consideration is included in this safety evaluation (Section 4.0).

[1] 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.

## 2.0 REGULATORY EVALUATION

### 2.1 Description of System

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, SG tube integrity means that no materials or other degradation would hinder the capability of tubes to perform this safety function in accordance with the plant design and licensing basis.

### 2.2 Regulatory Requirements and Guidance

Requirements for the integrity of the SG tubing are established in Title 10 of the *Code of Federal Regulations* (10 CFR). Specifically, General Design Criterion (GDC) 14, "Reactor coolant pressure boundary," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 states that the RCPB "shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture." GDC 15, "Reactor coolant system design," states that the reactor coolant system and associated auxiliary, control, and protection systems "shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences." GDC 31, "Fracture prevention of reactor coolant pressure boundary," states, in part, that the RCPB "shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized."

The regulations in 10 CFR 50.36, "Technical specifications," require that each applicant for an operating license includes (in its application) proposed TSs and establishes the regulatory requirements for the content of TSs. The TSs are required to be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34, "Contents of applications; technical information." The Commission may include such additional TSs as the Commission finds appropriate.

Given the importance of SG tube integrity, all current pressurized-water reactor (PWR) licensees have TSs governing the surveillance of SG tubes. The TSs for all PWR plants require 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 TSs. For Beaver Valley 1, the requirements for performing SG tube inspections and repair are in TS Section 5.5.5.1.

### 2.3 SG Tube Integrity Requirements in the Beaver Valley 1 TSs

At Beaver Valley 1, programs established by the licensee, including the SG Program, are listed in the administrative controls section of the TSs. The requirements for performing SG tube inspections and plugging are described in TS 5.5.5.1, while the requirements for reporting the SG tube inspections and plugging are described in TS 5.6.6.1, "Steam Generator Tube Inspection Report."

For Beaver Valley 1, SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.5.1.b for structural and leakage integrity. TS 5.5.5.1.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. TS 5.5.5.1.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspections be performed with the objective of detecting flaws of any type that may be present along the length of a tube and that may satisfy the applicable tube plugging criteria. The applicable tube plugging criterion specified in TS 5.5.5.1.c is that tubes found during inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged.

The Beaver Valley 1 TS 3.4.13, "RCS [Reactor Coolant System] Operational LEAKAGE," includes a limit on operational primary-to-secondary leakage beyond which the plant must be promptly shut down. Should a flaw exceeding the tube plugging limit not be detected during the periodic tube surveillance required by the plant TSs, the operational leakage limit provides added assurance of timely plant shutdown before tube structural and leakage integrity are impaired.

As part of the plant's licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents such as an SG tube rupture and a steam line break and include the results in the FSAR. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the radiological consequences do not exceed the applicable limits of 10 CFR 50.67, "Accident source term," or 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," for offsite doses; GDC 19, "Control room," of Appendix A to 10 CFR Part 50, for control room operator doses (or some fraction thereof, as appropriate to the accident); or the NRC-approved licensing basis (e.g., a small fraction of these limits). No accident analyses for Beaver Valley 1 are being changed because of the proposed amendment; thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the NRC has reviewed and approved for the postulated design-basis accidents for SG tubes.

## 3.0 TECHNICAL EVALUATION

### 3.1 Background

#### 3.1.1 SG Design

The original SGs in Beaver Valley 1 were replaced in spring 2006 during the 1R17 refueling outage. The replacement SGs are Westinghouse Electric Company LLC (Westinghouse) Model 54F SGs and are described in Section 4.2.2.4, "Steam Generators," of the Updated Final Safety Analysis Report (Reference 2). Each SG contains 3,592 thermally treated, Alloy 690, U-tubes with a nominal outer diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The tubes undergo thermal treatment following tube-forming and annealing operations and have been hydraulically expanded into the tubesheet. The tubes are supported by stainless steel tube support plates (TSPs) with broached quatrefoil holes, while the flow distribution baffle plate has octafoil shaped holes.

3.1.2 Operating Experience

Plugging History

Following the most recent inspection in 2016 (1R24), a total of one tube has been removed from service by plugging since the SGs were replaced in 2006, as shown below in Table 1.

**Table 1: Beaver Valley 1 – Tube Plugging Summary**

| Date | SG-A | SG-B | SG-C | Total |
|---|---|---|---|---|
| Pre-Service | 0 | 0 | 0 | 0 |
| 2007 (1R18) | 0 | 0 | 1 | 1 |
| 2012 (1R21) | 0 | 0 | 0 | 0 |
| 2016 (1R24) | 0 | 0 | 0 | 0 |
| **Total Tubes Plugged** | **0** | **0** | **1** | **1** |
| Total Percentage | 0.0% | 0.0% | 0.03% | 0.01% |

In 2007, during the first inspection following SG replacement (1R18), one tube in SG-C was plugged to address TSP wear. The wear indication was from a burr on the upper edge of the TSP and believed to be left from manufacturing.

Recent Inspection Summaries

The last two SG inspections at Beaver Valley 1 are summarized below. More information is available in the 1R21 (Reference 3) and 1R24 (Reference 4) SG tube inspection reports.

1R21 Summary

A single anti-vibration bar (AVB) wear indication with a depth of 9 percent through-wall (TW) in SG-A was detected and returned to service. The affected tube was left in service. Three TSP wear indications associated with burrs were detected in the three SGs, of which three were new and one was a repeat indication. The largest indication was 19 percent TW and all the affected tubes were left in service.

There were no foreign object (FO) wear indications reported but there were 68 FOs identified during the inspection. Sixty of the 68 FOs were successfully removed. The eight remaining FOs were determined to be acceptable for continued operation, since an engineering evaluation concluded tube integrity would be maintained for at least three more operating cycles.

The secondary-side inspection activities in all three SGs included tubesheet sludge lancing, tubesheet inspections, and foreign object search and retrieval. Channel head and plug visual inspections were also performed and no degradation was noted. Secondary-side visual inspections of the upper tube bundle were performed in SG C.

1R24 Summary

Eight AVB wear indications were reported in SGs A and B, of which seven were new and one was a repeat indication. The largest indication was 11 percent TW and all the affected tubes were left in service. Three historical TSP wear indications associated with burrs were reported,

one in each SG. The largest indication was 23 percent TW and all the affected tubes were left in service. There were 21 new TSP wear indications reported that were associated with TSP lands. The largest indication was 15 percent TW and all the affected tubes were left in service.

Sixteen FOs were observed on the top of the tubesheet during the inspection and all but four were successfully removed from the SGs. The remaining four FOs were determined to be acceptable for continued operation by an engineering evaluation. Three FO wear indications were detected during the inspections, one in SG-A and two in SG-B but eddy current inspection did not indicate the presence of FOs. The largest indication was 22 percent TW and all tubes were left in service.

The secondary-side inspection activities in all three SGs included tubesheet sludge lancing, tubesheet inspections, and foreign object search and retrieval. Channel head and plug visual inspections were also performed and no degradation was reported. Secondary-side visual inspections of the steam drum region were performed in SG-B and no degradation was reported.

Operational History Since 1R24

For Cycle 25, Beaver Valley 1 experienced a condenser tube leak in January 2018. The licensee reported that the responses and actions taken during the excursion met the time frames as required in the Electric Power Research Institute (EPRI) Final Report, "Pressurized Water Reactor Secondary Water Chemistry Guidelines," Revision 7 (Reference 5), which were in effect at that time, to return parameters to normal levels. A forced outage in November 2017 caused elevated SG sodium, chloride, and sulfate concentrations while the plant was shut down; however, no EPRI action levels were exceeded.

For Cycle 26, Beaver Valley 1 experienced a condenser tube leak in June 2018. The licensee reported that responses and actions taken during the excursion met the time frames as required in EPRI's Final Report, "Pressurized Water Reactor Secondary Water Chemistry Guidelines," Revision 8 (Reference 6), to return parameters to normal levels.

For Cycle 27, the licensee stated that no primary-to-secondary leakage has been reported and there have been no chemistry transients during the current operating cycle.

## 3.2 Proposed TS Changes

### 3.2.1 Current TS Requirements

The SG Program in Beaver Valley 1 TS 5.5.5.1 provides the SG tube inspection requirements. TS 5.5.5.1.d for Beaver Valley 1 requires periodic SG tube inspections to be performed and specifies provisions to be met for such inspections. TS 5.5.5.1.d.1 requires inspection of 100 percent of the tubes in each SG during the first refueling outage following SG installation. TS 5.5.5.1.d.2 states, "After the first refueling outage following SG installation, inspect each SG at least every 72 effective full-power months or at least every third refueling outage (whichever results in more frequent inspections)." Additionally, the subsections of TS 5.5.5.1.d.2 require 100 percent of the tubes to be inspected during sequential periods of 144, 120, 96, and 72 effective full-power months. For Beaver Valley 1, the 1R27 outage in the spring 2021 will be the first outage in the second inspection period, which is 120 effective full-power months in duration.

3.2.2 Description of Proposed TS Changes

The license amendment request proposes to annotate TS 5.5.5.1.d.2 with an asterisk at the end of the first sentence and a footnote with an asterisk at the bottom of the page would state:

> * The spring of 2021 (1R27) refueling outage SG inspections may be deferred to the fall of 2022 (1R28) refueling outage.

3.3 NRC Staff Evaluation of Proposed TS Changes

The NRC staff evaluation of the proposed one-time TS changes was performed within the context of the COVID-19 pandemic and the potential impacts of the COVID-19 virus on plant personnel. Therefore, this safety evaluation should not be considered precedent setting for future routine plant amendments or generic industry licensing actions related to SG inspection intervals.

The NRC staff evaluation of the proposed one-time TS changes focused on the potential for affecting SG tube integrity, since maintaining SG tube integrity ensures the plant will meet the SG Program related TS, thereby protecting public health and safety. In particular, the staff evaluation assessed whether the amendment request demonstrates that the structural integrity performance criterion (SIPC) and accident-induced leakage performance criterion (AILPC) will be met for Cycle 28. The SIPC and AILPC are defined in TS Section 5.5.5.1.b

Operational Assessment Methods

The only degradation mechanisms detected in the SG tubes at Beaver Valley 1 have been wear from AVBs, TSPs, and FOs.

The operational assessment (OA) provided by the licensee evaluates both detected and undetected forms of AVB and TSP wear using a deterministic analysis. These analyses use a single tube analysis method, to provide a conservative estimate of the projected end-of-cycle (EOC) condition considering all uncertainties at the upper 95$^{th}$ percentile. The applicable uncertainties are for the burst equation, the material strength, and the nondestructive examination (NDE) flaw sizing technique. The single tube methods are referred to as "worst-case degraded tube" methods because the most severely flawed tube is selected for evaluation at the beginning-of-cycle (BOC) under consideration, conservative flaw growth is applied, a predicted EOC flaw size is calculated, and a determination is made whether the SIPC and AILPC will be met at the EOC.

The OA also included a simplified Monte Carlo analysis for undetected AVB and TSP wear. This analysis projects the worst-case BOC flaw to the next scheduled inspection with a constant flaw growth and all the relevant uncertainties applied using at least 100,000 simulations. Each simulation randomly samples the standard normal deviate, Z, of each uncertainty term, from a normal distribution with a mean of zero and a standard deviation of 1. The relevant uncertainties sampled for each simulation using the Monte Carlo technique are tube material strength, burst relation, and NDE sizing uncertainties. Also, for this OA, a constant depth growth value distribution was conservatively used instead of the actual growth distribution. The proprietary (Westinghouse) flaw model software used calculates a burst pressure at 95 percent probability and 50 percent confidence level, at the end of the OA operating period for a given BOC flaw depth and BOC flaw length. This result is compared to the minimum required burst pressure necessary to maintain the SIPC, which is 4300 pounds per square inch (psi). All

inputs to this evaluation are the same as the deterministic evaluation described above, with the exception that the standard normal deviate, Z, is sampled, instead of using a constant upper 95$^{th}$ percentile value of 1.645.

While FO wear has occurred in the Beaver Valley 1 SGs, all known FOs causing wear were either removed from the SGs or the remaining FOs were evaluated and determined to not have potential to cause tube wear. Since there are no remaining FOs with the potential to cause wear or cause in-service tubes with indications of FO wear to grow, the licensee concluded the SG performance criteria for structural and leakage integrity will be satisfied until the EOC 28.

3.3.1 Evaluation of Existing Tube Degradation Mechanisms

Wear at AVBs

Wear at AVBs has been detected in two of the three SGs at Beaver Valley 1 but has not resulted in any tubes being plugged. In the 1R21 SG inspections, one AVB wear indication was detected. In the 1R24 SG inspections, eight AVB wear indications were detected in three tubes, seven indications in SG-A and one indication in SG-B. The 1R21 and 1R24 inspection scopes for AVB wear consisted of full-length bobbin probe examinations of 50 and 58 percent of the in-service tubes, respectively. Sizing of the AVB wear indications was based on the bobbin probe results using an EPRI-qualified technique. The deepest indication returned to service in 1R21 was 9 percent TW and in 1R24 it was 11 percent TW.

The licensee's OA for AVB wear was performed using the arithmetic approach described in the EPRI's "Steam Generator Integrity Assessment Guidelines" (EPRI SG IAG) (Reference 7). This approach adjusts the deepest AVB wear flaw returned to service to account for eddy current regression and sizing uncertainties, applies a 95$^{th}$ percentile growth rate, and then compares the projected flaw size at the next EOC inspection to the acceptable structural limits. Because the Beaver Valley 1 SGs have only one repeat AVB wear indication, out of a total population of only eight indications, a growth rate could not be determined. Instead, the licensee used growth rates from other similar SG models (Model 54F and Delta 54). The average growth rate of these SGs was 3 percent TW per effective full power year (TW/EFPY). A growth rate of 5 percent TW/EFPY was used for the Beaver Valley 1 SGs, as it was considered bounding and conservative, given the operating experience at Beaver Valley 1. The structural limit for AVB wear flaws is 52.1 percent TW, which includes burst strength and material property uncertainties. Since the EOC projected AVB wear flaw size of 41 percent TW is less than the structural limit for AVB wear flaws, the NRC staff concludes that the SIPC would be satisfied at the EOC 28.

Because the licensee inspected half of all the SG tubes in 1R21 and the other half in 1R24, the licensee assumed that an undetected AVB wear flaw was left in service in 1R21. To assess the possible size of this wear flaw, the licensee used the EPRI Model Assisted Probability of Detection Code (MAPOD) in the EPRI SG IAG to develop probability of detection curves for Beaver Valley 1. The licensee used the MAPOD Code to combine eddy current tube noise measurements from 1R21 and 1R24 with a bobbin probe detection voltage amplitude-to-true depth distribution correlation, to generate a site-specific noise-based probability of detection (POD) curve for maximum flaw depth. Based on the developed POD curves, the 95$^{th}$ percentile size of a missed AVB wear flaw during 1R21 was 7 percent TW. When projected to the EOC 28 with the 5 percent TW/EFPY growth rate used previously, the resulting EOC AVB wear flaw is 57 percent TW, which is less than the structural limit of 60 percent TW. The NRC staff notes that since the undetected flaw size is assumed and not measured, no NDE sizing uncertainty is

necessary. As noted in the EPRI guidance document, “Steam Generator Management Program: Steam Generator Degradation Specific Management Flaw Handbook” (EPRI SG Degradation Handbook) (Reference 8), wear flaws from AVBs will leak and burst at essentially the same pressure. Because the differential pressure limit for accident-induced leakage integrity is much lower than the differential pressure limit for structural integrity, by meeting the SIPC, the NRC staff concludes that the AILPC will also be satisfied.

The licensee also provided a simplified Monte Carlo analysis for a worst-case undetected AVB wear flaw of 8 percent TW, projected from 1R21 to 1R28 using the method described above in Section 3.3. The calculated burst pressure of 4465 psi at 0.95 probability and 50 percent confidence level compared favorably to the bounding SIPC limit of 4300 psi for structural burst and the structural limit of 60 percent TW, which also supports the NRC staff’s conclusion that the SIPC and AILPC will be satisfied.

Wear at TSPs

There are three TSP indications in three different tubes that the licensee has concluded were caused by tube contact with a metal burr on the edge of a TSP. The burrs were created during manufacture of the TSPs. When initially detected in 1R21, these three indications ranged in size from 15 to 19 percent TW. When measured again in 1R24, these indications showed low growth, with an average growth rate of about 1 percent TW/EFPY. The licensee concluded that the wear indications in the three tubes have reached a physical limit, based on the size of the three burrs and will experience no further growth. Based on growth rates calculated from eddy current sizing performed in 1R21 and 1R24, the NRC staff agrees that the three indications have seen a significant reduction in growth rate but the NRC staff cannot yet conclude that the TSP burr indications have stopped growing. Nevertheless, based on the measured sizes of the three indications in 1R24 and their growth rates shown in Table 2-5 of the license amendment request, the NRC staff concludes that there is reasonable confidence that these indications will not challenge tube integrity by the EOC 28.

Wear at TSPs (other than the three burr indications) has been detected in all three SGs at Beaver Valley 1 but has not resulted in any tubes being plugged. In the most recent 1R24 SG inspections, 21 new TSP wear indications were detected in 21 tubes. The 1R24 inspections for TSP wear consisted of full-length bobbin probe examinations of 58 percent of the active tubes. Sizing of the TSP wear indications was based on the bobbin probe results using an EPRI-qualified technique. The deepest indications returned to service in each SG were 15 percent TW in SG-A, 8 percent TW in SG-B, and 6 percent in SG-C.

The licensee’s OA for TSP wear was performed using the arithmetic approach described in the EPRI SG IAG. This approach adjusts the deepest TSP wear flaw returned to service to account for eddy current regression and sizing uncertainties, applies a 95$^{th}$ percentile growth rate, and then compares the projected flaw size at the next EOC inspection to the acceptable structural limits. The largest TSP growth rate calculated in the 1R24 OA was 4.3 percent TW/EFPY and was used to project TSP growth to the EOC 28. The structural limit for TSP wear flaws is 48.7 percent TW, which includes burst strength and material property uncertainties. Since the EOC projected TSP wear flaw size of 41 percent TW is less than the structural limit for TSP wear flaws, the NRC staff concludes that the SIPC would be satisfied at the EOC 28.

Because the licensee inspected half of all the SG tubes in 1R21 and the other half in 1R24, the licensee assumed that an undetected TSP wear flaw was left in service in 1R21. To assess the possible size of this wear flaw, the licensee used the EPRI MAPOD Code in the EPRI SG IAG

to develop probability of detection curves for Beaver Valley 1. The licensee used the MAPOD Code to combine eddy current tube noise measurements from 1R21 and 1R24 with a bobbin probe detection voltage amplitude-to-true depth distribution correlation, to generate a site-specific noise-based POD curve for maximum flaw depth. Based on the developed POD curves, the 95$^{th}$ percentile size of a missed TSP wear flaw during 1R21 was 8 percent TW. When projected to the EOC 28 with the 4.33 percent TW/EFPY growth rate used previously, the resulting EOC TSP wear flaw is 51 percent TW, which is less than the structural limit of 55 percent TW. The NRC staff notes that since the undetected flaw size is assumed and not measured, no NDE sizing uncertainty is necessary. As noted in the EPRI SG Degradation Handbook (Reference 8), wear flaws from TSPs will leak and burst at essentially the same pressure. Because the differential pressure limit for accident-induced leakage integrity is much lower than the differential pressure limit for structural integrity, by meeting the SIPC, the NRC staff concludes that the AILPC will also be satisfied.

The licensee also provided a simplified Monte Carlo analysis for a worst-case undetected TSP wear flaw of 8 percent TW, projected from 1R21 to 1R28 using the method described above in Section 3.3. The calculated burst pressure of 4530 psi at 0.95 probability and 50 percent confidence level compared favorably to the bounding SIPC loading limit of 4300 psi for structural burst and the structural limit of 55 percent TW, which also supports the NRC staff's conclusion that the SIPC and AILPC will be satisfied..

Evaluation Summary for Existing Mechanisms (Wear at AVBs and TSPs)

Wear at AVB and TSP locations in the Beaver Valley 1 SGs has been effectively managed since SG installation in 2006, without challenging tube integrity. Wear at support structures is readily detected with standard eddy current examination techniques and wear sizing errors are considered in the projection of existing flaws to EOC 28.

The licensee provided deterministic wear analyses for the AVBs and TSPs until 1R28. For AVB wear growth, the analysis used a conservative bounding growth rate based on industry operating experience and assumed a missed indication in 1R21. In a similar manner, the licensee evaluation of TSP wear used a conservative bounding growth rate based on Beaver Valley 1 operating experience and also assumed a missed indication in 1R21. The results of the AVB and TSP wear analyses, with projected conservative wear rates through 1R28, predict that tube integrity will be maintained. Therefore, the NRC staff finds the licensee's evaluation of wear at support structures to be acceptable since the SIPC and AILPC will be satisfied.

3.3.2 Evaluation of Potential Tube Degradation Mechanisms

Foreign Object Wear

The Beaver Valley 1 SGs have reported FO wear at the top of the tubesheet since SG installation in 2006. These instances of FO wear resulted in the plugging of no SG tubes. The 1R24 inspections for FO wear in each SG consisted of full-length bobbin probe examination of 58 percent of the active tubes in all SGs and +Point$^{TM}$ probe inspections of the tube bundle periphery, two to three tubes deep on both the hot and cold legs. Three indications of FO wear were detected at the top of tubesheet during the 1R24 inspections. Eddy current inspections did not indicate FOs remained at any of the three locations. Twelve of the 16 known FOs in the SGs were removed in 1R24 and the remaining four FOs were determined by engineering evaluation to not be a threat to tube integrity. Since there are no known remaining FOs with the potential to cause wear or cause in-service tubes with indications of FO wear to grow, the

licensee concluded the SG performance criteria for structural and leakage integrity will be satisfied until the EOC 28.

Evaluation Summary for Potential Mechanisms

The NRC staff finds the licensee's analysis of FO wear acceptable based on the licensee's analysis of the known FOs in the Beaver Valley 1 SGs. The staff acknowledges that predicting future FO generation is not possible, since past fleet-wide operating experience has shown that new FO generation, transport to the SG tube bundle, and interactions with the tubes cannot be reliably predicted. However, plants can reduce the probability of FOs by maintaining robust foreign material exclusion programs and applying lessons learned from previous industry operating experience with FOs. Plants in general, have demonstrated the ability to conservatively manage FOs once they are detected by eddy current examinations or by secondary-side foreign object search and retrieval inspections. If unanticipated aggressive tube wear from new FOs should occur in a Beaver Valley 1 SG, operating experience has shown that a primary-to-secondary leak will probably occur, rather than a loss of tube integrity. In the event of a primary-to-secondary leak, the staff will interact with the licensee in accordance with established procedures in Inspection Manual Chapter (IMC) 0327, "Steam Generator Tube Primary-to-Secondary Leakage," dated January 1, 2019 (Reference 9), to confirm the licensee's conservative decision making.

## 3.4 Primary-to-Secondary Leakage Actions

Beaver Valley 1 operated with no detectable primary-to-secondary leakage from the time of SG replacement in 2006. TS 3.4.13, "RCS Operational LEAKAGE," has a limiting condition for operation that requires reactor coolant system operational leakage to be limited to ≤150 gallons per day through any one SG while operating in Modes 1 through 4. In addition to the TS requirements, Beaver Valley 1 has administrative limits for responding to primary-to-secondary leakage during operation. These limits require increased levels of monitoring starting at leakage of 5 gallons per day or more. The licensee proposed no changes to these existing TS and administrative limits. The NRC staff finds this acceptable since the administrative limits require prompt and controlled shut down at a significantly lower primary-to-secondary leakage level compared to the TS Operational Leakage limits.

## 3.5 Technical Evaluation Conclusion

Based on the above, the NRC staff finds that the licensee has demonstrated there is reasonable assurance that the structural and leakage integrity of the Beaver Valley 1 SG tubes will be maintained until the next SG tube inspections during 1R28 in the fall of 2022. Therefore, the NRC staff concludes that the licensee may incorporate the proposed changes into TS 5.5.5.1.

# 4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change adds a note to TS 5.5.5.1.d.2 to permit a one-time deferral of the SG inspections from the spring of 2021 (1R27) refueling outage to the fall of 2022 (1R28) refueling outage. An operational assessment has been performed that concludes the SGs will continue to meet the structural and leakage integrity performance criteria throughout the proposed operating period, ensuring there is no significant increase in the probability of a previously-evaluated accident. The existing TS limit for addressing potential primary-to-secondary leakage is not altered, ensuring accident analysis initial assumptions are met and that there should be no significant increase in the consequences of a previously-evaluated accident.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously-evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change adds a note to TS 5.5.5.1.d.2 to permit a one-time deferral of the SG inspections from the spring of 2021 (1R27) refueling outage to the fall of 2022 (1R28) refueling outage. An operational assessment has been performed that concludes the SGs will continue to meet the structural and leakage integrity performance criteria throughout the proposed operating period. Furthermore, there are no physical system, structure, or component changes that could create the possibility of a new or different kind of accident.

   Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change adds a note to TS 5.5.5.1.d.2 to permit a one-time deferral of the SG inspections from the spring of 2021 (1R27) refueling outage to the fall of 2022 (1R28) refueling outage. An operational assessment has been performed that concludes the SGs will continue to meet the structural and leakage integrity performance criteria throughout the proposed operating period. Since the proposed inspection deferral

does not exceed or alter a design basis or safety limit, it does not significantly reduce the margin of safety.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff reviewed the licensee's no significant hazards consideration analysis. Based on the review and on the NRC staff's evaluation of the underlying license amendment request as discussed above, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

## 5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Commonwealth of Pennsylvania official was notified of the proposed issuance of the amendment on March 1, 2021. The Commonwealth official had no comments.

## 6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component 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 (86 FR 9087; February 11, 2021). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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.

## 7.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

## 8.0 REFERENCES

1. Beaver Valley Power Station, Unit No. 1, License Amendment Request for One-time Deferral of Steam Generator Inspections, dated January 27, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21027A228).

2. Beaver Valley Power Station, Unit 1, Revision 32 to Updated Final Safety Analysis Report, Chapter 4, "Reactor Coolant System," dated May 20, 2020 (ADAMS Accession No. ML20160A063).

3. Beaver Valley Power Station Unit 1, 2012 Steam Generator Tube Inspection Report, dated October 24, 2012 (ADAMS Accession No. ML12299A088).

4. Beaver Valley Power Station Unit 1, 2016 Steam Generator Tube Inspection Report, dated February 13, 2017 (ADAMS Accession No. ML17044A360).

5. Electric Power Research Institute, “Pressurized Water Reactor Secondary Water Chemistry Guidelines,” Final Report, Revision 7, February 2009 (ADAMS Accession No. ML11220A116).

6. Electric Power Research Institute, “Pressurized Water Reactor Secondary Water Chemistry Guidelines,” Final Report, Revision 8, September 2017 (ADAMS Accession No. ML19044A529; not publicly available).

7. Electric Power Research Institute, “Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines,” Revision 4, June 2016 (ADAMS Accession No. ML16208A273; not publicly available).

8. Electric Power Research Institute, “Steam Generator Management Program: Steam Generator Degradation Specific Management Flaw Handbook,” Revision 2, October 2015 (ADAMS Accession No. ML17103A395; not publicly available).

9. Inspection Manual Chapter (IMC) 0327, “Steam Generator Tube Primary-to-Secondary Leakage,” dated January 1, 2019 (ADAMS Accession No. ML18093B067).

Principal Contributor: A. Johnson

Date: March 22, 2021

SUBJECT: BEAVER VALLEY POWER STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT NO. 310 RE: ONE-TIME EXTENSION OF UNIT NO. 1 STEAM GENERATOR INSPECTIONS [COVID-19] (EPID L-2021-LLL-0002) DATED MARCH 22, 2021

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