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

February 25, 2019

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

SUBJECT: BEAVER VALLEY POWER STATION, UNIT 2 – ISSUANCE OF AMENDMENT NO. 193 RE: REVISE STEAM GENERATOR TECHNICAL SPECIFICATIONS (EPID L-2018-LLA-0075)

Dear Mr. Bologna:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 193 to Renewed Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit 2. This amendment consists of changes to the technical specifications in response to your application dated March 28, 2018, as supplemented by letter dated October 28, 2018.

The amendment revises various technical specification sections associated with steam generators to allow the use of Westinghouse leak-limiting Alloy 800 sleeves for an additional three fuel cycles of operation, bringing the total usage time from five to eight fuel cycles of operation. The technical specification changes also clarify wording in two sections related to use of leak-limiting Alloy 800 sleeves.

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

Sincerely,

Jenny Tobin for

Carleen J. Parker, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosures:
1. Amendment No. 193 to NPF-73
2. Safety Evaluation

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION, LLC

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 193
Renewed License No. NPF-73

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, (FENOC)* acting on its own behalf and as agent for FirstEnergy Nuclear Generation, LLC (the licensees), dated March 28, 2018, as supplemented by letter dated October 28, 2018, 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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*FENOC is authorized to act as agent for FirstEnergy 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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 193, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

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

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

Date of Issuance: February 25, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 193

BEAVER VALLEY POWER STATION, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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 a marginal line indicating the area of change.

Remove	Insert
Page 4	Page 4

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

Remove	Insert
5.5-12	5.5-12

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

FENOC 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. 193, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

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 steam generator 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. 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

   Steam generator 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. An Alloy 800 sleeve installed in the hot-leg or cold-leg tubesheet region shall remain in service for no more than eight fuel cycles of operation starting from the outage when the sleeve was installed.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 193 TO RENEWED

FACILITY OPERATING LICENSE NO. NPF-73

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRST ENERGY NUCLEAR GENERATION, LLC

BEAVER VALLEY POWER STATION, UNIT 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By application dated March 28, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18087A293), as supplemented by letter dated October 28, 2018 (ADAMS Accession No. ML18302A244), FirstEnergy Nuclear Operating Company (the licensee), requested changes to the technical specifications (TSs) for the Beaver Valley Power Station (Beaver Valley), Unit 2.

The proposed changes would revise various sections of the TSs associated with steam generators, including increasing the number of fuel cycles that Westinghouse Electric Company LLC leak-limiting Alloy 800 sleeves may remain in operation from five to eight fuel cycles of operation. The proposed TS changes also clarify wording in two sections related to use of the leak-limiting Alloy 800 sleeves.

The supplement dated October 28, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 5, 2018 (83 FR 26105).

2.0 REGULATORY EVALUATION

The licensee requested a change to the renewed facility operating license for Beaver Valley, Unit 2, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90. The regulations in Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC), to 10 CFR Part 50, or similar plant-specific principal design criteria, provide design requirements. The regulatory requirements of 10 CFR Part 50, Appendix A, that are applicable to the reactor coolant pressure boundary (RCPB) include GDC 14, 15, 30, 31, and 32. GDC 14 requires that the subject systems shall have "an extremely low probability of abnormal leakage ... and gross rupture." GDC 15 and 31 require that the RCPB "shall be designed with sufficient margin."

GDC 30 requires that the RCPB shall be "the highest quality standards possible." GDC 32 requires that the RCPB shall be designed to permit "periodic inspection and testing ... to assess ... structural and leak tight integrity."

Beaver Valley, Unit 2, was evaluated against the GDC requirements discussed above. The Beaver Valley, Unit 2, construction permit was issued in May 1974, and conforms with the standards set forth in the GDC.

As specified in 10 CFR 50.55a, components that are part of the RCPB must meet the requirements of Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Section 50.55a of 10 CFR further requires that throughout the service life of a pressurized-water reactor (PWR) facility, ASME Code Class 1 components meet the requirements in Section XI of the ASME Code to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to inservice inspection of steam generator (SG) tubing are augmented by additional SG tube surveillance requirements in the TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents, such as an SG tube rupture and main steamline break. These analyses consider the primary-to-secondary leakage through the tubing that may occur during these events. Furthermore, the analyses must show that the offsite radiological consequences do not exceed the applicable limits of 10 CFR Part 100 guidelines for offsite doses (or 10 CFR 50.67, as appropriate); GDC 19, "Control Room," criteria for control room operator doses, or some fraction thereof, as appropriate to the accident; or the NRC-approved licensing basis.

In Section 50.36 of 10 CFR, the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operating (LCOs); (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in the plant's TSs.

Under 10 CFR 50.36(c)(5), administrative controls are defined as "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." Programs established by the licensee to operate the facility in a safe manner, including the SG program, are listed in the administrative controls section of the TSs. The SG program is defined in TS 5.5.5, "Steam Generator (SG) Program."

TS 5.5.5.2 for Beaver Valley, Unit 2, requires that an SG program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria for structural and leakage integrity consistent with the plant design and licensing bases. TS 5.5.5.2.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. TS 5.5.5.2.f includes provisions regarding the scope, frequency, and methods of SG tube repair.

3.0 TECHNICAL EVALUATION

The three SGs at Beaver Valley, Unit 2, are Westinghouse Model 51M SGs. Each SG contains 3,376 mill-annealed Alloy 600 tubes. Each tube has a nominal outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The tubes are supported by a number of carbon steel tube support plates (TSPs) and Alloy 600 anti-vibration bars. The tubes were roll expanded at both ends for the full depth of the tubesheet. The portion of tubes from about 3 inches above the top of the tubesheet to about 1 inch above the tube ends was shot-peened (on both the hot leg and cold leg side of the SG), prior to operation. In addition, the U bend region of the small radius tubes were in-situ stress relieved prior to operation. The design, installation, analysis, and qualification tests of these sleeves were previously submitted in the Westinghouse topical report, WCAP-15919-NP, Revision 2, "Steam Generator Tube Repair for Westinghouse Designed Plants with 7/8 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," dated January 2006 (ADAMS Accession No. ML082890824; non-proprietary version).

3.1 Description of Proposed Changes

Since the NRC staff previously approved this sleeve design for five cycles of operation at Beaver Valley, Unit 2, the staff focused its review on development of the technical basis for extending the approval for an additional three cycles of operation. In Section 4.0 of its license amendment request (LAR), the licensee stated that the overall logic for its evaluation in extending sleeve service life from five to eight cycles includes:

1. Residual stresses of the parent tube inside diameter (ID) surface are compressive, and thus, stress corrosion cracking (SCC) of the parent tube will not occur.

2. If degradation of the parent tube were to occur, current eddy current inspection capabilities have advanced to the point where they are capable of ensuring that part through-wall (TW) degradation of the parent tube behind the nickel band is readily detectable.

3. Mechanical load testing shows that SCC of the parent tube adjacent to the sleeve's nickel band does not reduce the axial load bearing capability of the tube below the design requirement.

The NRC staff also reviewed the proposed wording clarifications to Beaver Valley, Unit 2, TSs 5.5.5.2.d.5 and 5.5.5.2.f.3, to determine the acceptability of the proposed changes.

Sleeve Design

A sleeve is a tube segment that is inserted into an existing SG tube and expanded at both ends of the sleeve to form a structural joint. The leak-limiting sleeve is not designed to be leaktight. Two leak-limiting Alloy 800 sleeve designs were proposed for use in repairing SG tubes, a transition zone (TZ) sleeve, and a tube support sleeve. The TZ sleeve is designed for tube degradation near the top of the tubesheet. The upper end of the TZ sleeve forms a sleeve-to-tube joint by six equally spaced hydraulic expansions. The bottom end of the TZ sleeve forms a sleeve-to-tube joint by roll expansion of the sleeve into the tube within the tubesheet and includes both a nickel band and a thermally-sprayed nickel alloy band on the outside diameter surface of the sleeve. The function of the nickel band is to improve the sealing of the joint, while the function of the thermally-sprayed nickel alloy "microlok" band is to increase

the strength of the joint. The tube support sleeve is designed for tube degradation at TSP intersections or in the freespan region of SG tubes. The length of the TZ and tube support sleeves are sized according to the length of the degraded tubing regions into which they are inserted. Currently, Beaver Valley, Unit 2, only has TZ sleeves installed in its SGs.

Sleeve Installation

The licensee stated that the leak-limiting Alloy 800 sleeves will be installed in accordance with the processes provided by the vendor and described in the associated reports, which address sleeve design, qualification, installation methods, non-destructive examination, and as low as reasonably achievable radiation dose considerations. Installation of the sleeves will conform to ASME Code, Section XI, IWA-4720.

In the initial NRC approval for Beaver Valley, Unit 2, to use the leak-limiting Alloy 800 sleeves, the installation process included a step where the inside surface of a candidate tube was mechanically conditioned with a high-speed buffing tool. The initial LAR also included a statement that the buffing process might be eliminated in the future when sufficient confidence had been established in the process. In a Category 1 public meeting with the NRC on December 5, 2017 (ADAMS Accession No. ML17347A808), the licensee confirmed that the buffing process had been eliminated as a standard part of the installation process since it was not required for creating an acceptable sleeve-to-tube joint.

The sleeve is mounted on an expansion device and inserted into a tube for expansion. The expansion device is controlled and monitored to ensure consistent diametrical expansion. A hydraulic expansion tool is used at both ends of the tube support sleeves and at the top end of the TZ sleeve, while a roll expander is used at the bottom end of the TZ sleeve. The sleeve-to-tube joint formed by the hydraulic expansion generates the required structural and leakage integrity, while limiting the residual stresses in the parent tube. The torque of the roll expander is also monitored and controlled during installation. After the installation, all sleeve-to-tube joints undergo an initial acceptance and baseline inspection using an eddy current technique.

Sleeve Material Selection

The sleeve material, Alloy 800, is a nickel-iron-chromium alloy that was selected for its favorable properties, including corrosion resistance in both the primary and secondary side water chemistries. The Alloy 800 material is procured in accordance with the requirements of ASME Code, Section II, Part B, SB-163, NiFeCr Alloy, Unified Numbering System N08800, and Section III, Subsection NB-2000. There are additional content restrictions on various chemical elements and a specific annealing temperature and yield strength for the Alloy 800 sleeve. This material is acceptable to the NRC staff since it is allowed by the ASME Code, which the NRC has incorporated by reference in 10 CFR 50.55a.

Sleeve Qualification Testing

Prior to the original installation of these sleeves in the Beaver Valley, Unit 2, SGs, the licensee performed qualification tests in accordance with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. The testing program included mechanical load tests, leakage tests, and corrosion tests. The mechanical load tests included axial load, pressure, collapse, and load cycling. The tests were performed on

sleeve/tube mock-ups that were constructed to the same dimensions as the installed sleeves in the field.

Mechanical Testing

Axial load tests were performed to determine the structural integrity of the sleeve/tube joint. Axial loads are imposed as a result of the different thermal expansion rates of the leak-limiting Alloy 800 sleeve, as compared to the Alloy 600 tube, and due to the differential pressure across the walls of the sleeve and tube. The test loads included the full range of loadings expected under transient, normal power, and accident conditions. The axial load tests showed that the leak-limiting Alloy 800 sleeve experiences only minor displacement, even when the parent tube is severed, and will not result in tube-to-tube contact in the U-bend area.

Collapse tests were also performed to show that the sleeve would not collapse following a loss-of-coolant accident (LOCA). The collapse tests showed that the sleeve would not collapse, even at secondary to primary differential pressures well above those experienced following a LOCA. The tests showed that once the pressure got high enough in the gap between the tube and the sleeve, the pressure would vent through the joint.

Load-cycling tests were performed to show that the structural and leakage integrity of the sleeve/tube joint will be maintained under cyclical, differential thermal expansion and internal pressure in normal operating and transient conditions. The load-cycling tests included fatigue, thermal cycling, and mechanical load cycling. The load applied in the cycling tests was greater than three times the maximum operating differential pressure load. These tests showed that under various temperatures, the sleeve/tube joint is not significantly degraded by cyclic loads. The cycling tests confirm that slip during the initial heatup is small, and the sleeve repositions itself inside of the parent tube to accommodate the thermal expansion without subsequent slip. As a part of the load-cycling tests, the specimens were also tested for leakage integrity. The leak tests showed that the seal in the hydraulically expanded joints improved after load cycling.

Leak-rate tests were performed on the sleeve/tube assembly for various temperatures and pressures under normal operating and main steam line break conditions. The test results showed that the leakage from a single sleeve is extremely small relative to both the operational primary-to-secondary leakage limit in the plant TSs and the allowable leakage under accident conditions (see further discussion in Section 3.6 below).

The NRC staff finds the mechanical testing acceptable since it was performed under a quality assurance program, and it verified that the load carrying capability of the sleeve/tube assembly met the regulatory acceptance criteria for structural and leakage integrity discussed above in Section 2.

Corrosion Testing

Various corrosion tests were performed, including assessments of the leak-limiting Alloy 800 sleeves with full-length sleeved tube mock-ups. Sleeve/tube assemblies were pressurized with highly corrosive solutions and corrosion tests to assess the relative time to cracking of the sleeve/tube joint were also performed. The leak-limiting Alloy 800 sleeves did not develop any cracking in either the primary or secondary side tests, and the leak-limiting Alloy 800 sleeve demonstrated higher corrosion resistance than the Alloy 600 parent tube. The licensee stated that the leak-limiting Alloy 800 sleeves have not experienced service-induced degradation or leakage in nuclear plants. The licensee also stated that besides leak-limiting Alloy 800 sleeves,

Alloy 800 tubing has been used in PWR conditions in international nuclear plants with excellent results, based on the experience of over 200,000 tubes in service.

In the previous NRC approval of leak-limiting Alloy 800 sleeves at Beaver Valley, Unit 2, dated December 16, 2015 (ADAMS Accession No. ML15294A439), the staff noted that some indications of potential SCC were found in Alloy 800 tubing (not sleeves) in international plants. The staff also noted that the time for the initiation of corrosion in sleeve/tube assemblies was difficult to quantify accurately and that the accelerated lab testing method was unreliable for deterministic predictions. While the staff did consider the corrosion tests to give a viable indicator of potential performance, at that time, the staff assumed a limited life expectancy for leak-limiting Alloy 800 sleeves. Since the previous approval of the leak-limiting Alloy 800 sleeves at Beaver Valley, Unit 2, a few additional indications of outside diameter stress corrosion cracking (ODSCC) have been found in Alloy 800 tubing (not sleeves) in international plants.

In the current LAR, the licensee included an additional Westinghouse document (LTR-SGMP-18-3, dated March 28, 2018, ADAMS Accession No. ML18087A293), as a technical basis for requesting the extension of three additional operating cycles. The Westinghouse report provided evidence to show that the tensile stress component of SCC likely does not exist in the parent tubing beneath the bottom roll joint at Beaver Valley, Unit 2. For SCC to occur, the three elements of 1) a susceptible material, 2) a conducive environment, and 3) a residual tensile stress, are required. The licensee provided both operating experience and analytical evidence to support its claim that SCC is unlikely to occur beneath the bottom roll joint. Although the licensee submitted information that provides the staff additional confidence in the corrosion resistance of the parent tube underneath the bottom roll joint, there is still some uncertainty in the long-term corrosion behavior of the sleeve and parent tube at the joints. Sleeves are inspected with a specialized eddy current probe each refueling outage to ensure that flaws in the sleeve/tube are detected and addressed. The staff finds this acceptable, since it should ensure the timely detection of any degradation, should it occur.

3.2 Operating Experience

To address the issue of potential SCC in the leak-limiting Alloy 800 sleeves, the licensee provided additional operating experience data and analytical evidence, as discussed below.

In the current LAR, the licensee provided additional documentation of leak-limiting Alloy 800 sleeve operational experience, which showed that as many as 8,750 of these sleeves were in service at one time, with no reports of parent tube degradation. In total, more than 14,000 leak-limiting Alloy 800 TZ sleeves have been installed worldwide, plus approximately 12,000 additional tube support sleeves.

As noted previously, the SG tubes at Beaver Valley, Unit 2, were shot-peened prior to operation to induce a compressive residual stress on the ID surface of the SG tubes from 3 inches above the top of the tubesheet to 1 inch from the tube-end. Creating a compressive residual stress (as opposed to a tensile residual stress) on the ID surface of the tubes makes the tubes less susceptible to primary water stress corrosion cracking (PWSCC). The licensee's technical basis document shows only six PWSCC indications have been detected at Beaver Valley, Unit 2, all at the top of the expansion transition, the area typically of the highest residual stress. By comparison, two domestic plants with similar SGs and operating experience had 833 and 2,313 SG tubes with PWSCC indications. To date, only three of the sleeved tubes at Beaver Valley, Unit 2, have been removed from service through plugging of the parent tube, and none of these

removals were due to degradation of the parent tube within the tubesheet, which is the subject of this safety evaluation.

The licensee also provided a comparison of the operating experience from plants with mill-annealed SG tubing that was explosively expanded, hydraulically expanded, and mechanically roll expanded. In all cases, it was shown that the incidence of PWSCC is highest at the top of the tubesheet and decreases significantly with depth into the tubesheet. This finding is very relevant for TZ sleeves, as they are roll expanded into the middle portion of the tubesheet.

In this LAR, the licensee noted operational experience with over 9,500 laser welded and tungsten inert gas welded sleeves at Beaver Valley, Unit 1; Braidwood Station, Unit 1; Byron Station, Unit No. 1; and Joseph M. Farley Nuclear Plant, Units 1 and 2. Based on the observed flaw initiations of SG tubes in Farley Nuclear Plant, Unit 2 (in un-sleeved SG tubes), a Weibull initiation model predicted that the parent tubes of some sleeved SG tubes should have experienced degradation after sleeves had been installed. Since none of the 1,171 sleeved parent tubes showed indications of degradation after installation, the licensee concluded that either the residual stresses of the parent tube ID did not support SCC initiation, the installation of the sleeves resulted in a reduced tube temperature that increased the time to SCC initiation, or both.

In the supplement dated October 28, 2018, the licensee's response to NRC staff questions provided additional information that showed hundreds of SG tubes that contained PLUSS (PLUg replacing Sleeve which also Stabilizes) sleeves have been in operation for over 12 effective full power years at two international units. The PLUSS sleeves are identical to the Alloy 800 leak-limiting sleeves at Beaver Valley, Unit 2, except that there is no nickel band on the PLUSS sleeves. The NRC staff found this information to be valuable because this international operating experience bounds the number of effective full power years that the Beaver Valley, Unit 2, sleeves will have at the end of the proposed sleeving extension.

3.3 Analytical Evidence

Through the use of ANSYS (fluent software) modeling of the tubesheet, tube, and sleeve, the tube ID temperature where the tube was in intimate contact with the sleeve was determined. The result of the modeling showed that the sleeve will effectively reduce the temperature of the tube at the roll joint. Using Weibull initiation function analysis, combined with the temperature adjustment factor determined by the Arrhenius equation, it was shown that the decrease in temperature resulted in an increased characteristic life of the tube population of about 32 effective full power years.

A reference was provided in the technical basis document of finite element modeling papers,[1] which showed that the residual stresses on the SG tube ID surface remain compressive adjacent to the nickel band and within the length of the sleeve that is compressed by the flat length of the roller pin. This analysis was performed with two finite element analyses and one theoretical incremental analysis, and all three analyses showed similar results.

The NRC staff reviewed the operating experience and analytical evidence provided in the LAR and technical basis document. Although the staff did not perform a detailed review of the finite

[1] "Residual stresses associated with the hydraulic expansion of steam generator tubing into tubesheets," Nuclear Engineering and Design Volume 143, September 1993.

element analysis, the staff finds that the analysis results and the additional successful operating experience provide further support for three additional cycles of operation.

Sleeve Inspection

As part of the 2009 LAR, the licensee proposed a TS requirement to perform an inspection of the parent tube in the area where the sleeve-to-tube hard roll joint (lower joint) and the sleeve-to-tube hydraulic expansion joint (upper joint), will be established prior to installation of the sleeve. Sleeve installation will proceed only if the inspection finds these regions free from service-induced indications. This examination would ensure the area where the joints are to be established are free of detectable flaws, which provides additional assurance against degradation that could lead to leakage or compromise the integrity of the sleeve-to-tube joint.

To ensure effective inspections of the sleeve/tube assembly could be performed, the capability to inspect these regions was assessed. The qualification program included fabricating samples with axially and circumferentially oriented notches representing flaws at each of the transition and expansion zones. In addition, flaws in the pressure boundary portion of the sleeve and the parent tube away from the expansion regions were included in the sample set. The flaws included electro-discharge machined notches and a limited number of samples with cracking in the parent tube.

The original qualification program (prior to 2008) did not include flaws in the parent tube behind the nickel band portion of the Alloy 800 sleeve. Because of NRC staff questions pertaining to the inspection of this region of the sleeve/tube assembly, assessments of the capability to inspect this region were performed, and the consequences of having undetected flaws in this region were assessed. Attached to the October 10, 2008, LAR application, the licensee included a Westinghouse letter FENOC-08-148, "Summary of Alloy 800 Sleeve Parent Tube Eddy Current Test Results," dated September 26, 2008, which described test results related to the ability to detect flaws in the parent tube adjacent to the nickel band in tubes repaired using Alloy 800 sleeves. The report concluded that eddy current examinations could detect outside diameter electro-discharge machined flaws with depths ranging from 40 to 100 percent TW in the parent tube behind the nickel band of the Alloy 800 sleeves. The report also concluded that ODSCC flaws that are 100 percent TW, and approaching 100 percent TW, were readily detectable at this location using current examination techniques. The licensee also provided data assessing the strength of the sleeve/tube assembly if the portion of the parent tube behind the nickel band was not present. This data suggested that the sleeve joint would continue to have adequate axial load carrying capability.

In the October 28, 2018, supplement, the licensee's response to NRC staff questions provided additional information about the inspection process used on the installed sleeves that had been in operation in both domestic and international units. The NRC staff found the response acceptable because it clarified the source of the correlation between eddy current signal and +Point™ probe amplitude response, clarified how signal noise in the nickel band was addressed in a conservative manner in the inspection process, and provided evidence of flaw detection at alternate frequencies.

In the technical basis document dated March 10, 2018, in the March 28, 2018, LAR, the lack of a direct amplitude correlation between the 300 kilohertz (kHz) (parent tube) and 70 kHz (tube/sleeve combination) inspection frequencies was noted as a challenge to performing the model-assisted probability of detection (MAPOD) simulation. MAPOD is a Monte Carlo based methodology that randomly samples from an eddy current noise distribution and from a flaw's

eddy current signal-to-depth relationship, which is called the A-hat function. While performance assessments based on the MAPOD simulations could benefit from closer correlations between the A-hat function and PWSCC in the parent tube behind the Ni band (and a more conservative estimate of noise distribution in the region of interest), the probability of detection simulation results presented in the March 10, 2018, document are generally in agreement with the results from limited experimental assessments using nondestructive evaluation specimens with laboratory produced ODSCC. Although the NRC staff has remaining questions about the uncertainties associated with the MAPOD simulation performed by the licensee, the NRC staff approves the LAR due to its time-limited nature and the additional information provided in the October 28, 2018, supplement. The staff notes that a qualified inspection technique would be needed for approval of the leak-limiting Alloy 800 TZ sleeves on a permanent basis.

3.4 Sleeve Structural Analysis

Structural analyses were performed in accordance with 10 CFR Part 50, Appendix B, and Section III of the ASME Code. The structural analyses included applied loads under normal and accident loading conditions. The analyses assumed two bounding tube configurations: (1) the tube is intact, and (2) the tube is severed at the flaw location. In addition, the analyses assumed two bounding TSP configurations: (1) the tube is free to move past the TSPs, and (2) the tube is locked in the first TSP and is prevented from axial motion. The structural analyses showed that stresses and fatigue factors in the worst sleeve/tube configuration satisfied the allowable stress and fatigue factor values in Section III of the ASME Code.

The structural analyses also included calculations for a minimum required sleeve thickness based on the ASME Code, Section III. The calculations showed that the actual sleeve wall thickness is greater than the minimum required thickness, and therefore, is acceptable. The percentage of sleeve wall thickness that could be acceptably degraded was also calculated. This calculation considered axial and circumferential cracking. The calculated amount of degradation that could be tolerated and still meet ASME limits was considered acceptable to the NRC, since degradation of the sleeve is unlikely for the period of time the sleeve will be inservice (less than 8 years), and the licensee will plug all flaws on detection.

The NRC staff finds that the licensee's structural analysis is consistent with the ASME Code, and is, therefore, acceptable.

Under severe accident conditions in which primary system temperature may reach 1200 degrees Fahrenheit (°F) to 1500 °F, the material properties of Alloy 800 are not significantly different from that of Alloy 600. As a result, the structural integrity of the leak-limiting Alloy 800 sleeve is commensurate with the integrity of the Alloy 600 parent tubing under severe accident conditions, which is acceptable to the NRC staff, since the overall behavior of the sleeve/tube assembly will not change.

3.5 Sleeve Leakage Integrity

The sleeve-to-tube joint leakage was determined by laboratory testing to be small. For the leakage integrity assessment methodology, the licensee will conservatively assume all installed sleeves will leak under post-accident leakage conditions. The leak rate for each sleeve is an upper 95 percent confidence limit on the mean value of leakage for the appropriate temperature and pressure conditions. The licensee will combine the total sleeve leak rate with the total amount of leakage from all other sources (i.e., alternate repair criteria and non-alternate repair criteria indications) for comparison against the limit on accident-induced leakage, as specified in

the Updated Final Safety Analysis Report for all design-basis accidents. The staff finds that the licensee's leakage integrity assessment methodology is acceptable, since it assumes a conservative estimate of leakage from each sleeve, combines this estimate with the leakage from all other sources, and then compares the combined value against the acceptance limits.

3.6 Technical Specification Changes

The licensee stated it would delete "(lower joint)" and "(upper joint)" from TS 5.5.5.2.d.5 so it is clear that the inspection description was applicable to both types of sleeves, regardless of sleeve location. This change clarifies the specification without changing the original intent, which the NRC staff finds acceptable.

The licensee also proposed to add "installed in the hot-leg or cold-leg tubesheet region" in TS 5.5.5.2.f.3, which would clarify that the specification only applies to TZ sleeves installed in the tubesheet. This change is acceptable to the NRC staff since it clarifies that the service life limitation of eight fuel cycles is only applicable to the TZ sleeves, and not the tube support sleeves. Although Beaver Valley, Unit 2, does not have tube support sleeves, as noted by licensee in the LAR, because the tube support sleeves do not have a nickel band, non-destructive examination methods have been demonstrated to be effective in accordance with the requirements of the Electric Power Research Institute SG examination guidelines, and therefore, a limit on tube support sleeve operating life is not necessary.

The licensee also proposed extending the service life of TZ sleeves from five to eight fuel cycles, as documented in the technical basis provided with the LAR. As discussed above in Sections 3.1 through 3.5, the NRC staff has reviewed the technical basis provided for the service life extension from five to eight fuel cycles and finds it acceptable.

The licensee's proposed TS revisions are shown below. Additions are noted in **bold** and deletions with a ~~strikethrough~~.

5.5.5.2.d. Provisions for SG Tube Inspections

5. For Alloy 800 sleeves: The parent tube, in the area where the sleeve-to-tube hard roll joint ~~(lower joint)~~ and the sleeve-to-tube hydraulic expansion joint ~~(upper 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.

5.5.5.2.f. Provisions for SG Tube Repair Methods

3. Westinghouse leak-limiting Alloy 800 sleeves, WCAP-15919-P, Revision 2. An Alloy 800 sleeve **installed in the hot-leg or cold-leg tubesheet region** shall remain in service for no more than ~~five~~ **eight** fuel cycles of operation starting from the outage when the sleeve was installed.

Since these proposed changes are consistent with the technical basis evaluated above, these proposed TS changes are acceptable to the NRC staff.

3.7 Technical Evaluation Conclusion

Based on the information provided in the LAR, the NRC staff finds the licensee's inspection program acceptable, since the licensee (a) will be inspecting the parent tube at the location where the sleeve joints will be established to ensure the region is free of detectable flaws prior to sleeving, (b) has demonstrated that severe degradation in the joints can be detected, (c) has determined that the axial load carrying capability of the joint is not compromised in the event that severe degradation is present behind the nickel band region of the Alloy 800 sleeve, (d) has provided evidence that the mechanical roll portion of the sleeve will effectively remove the parent tube from the conducive environment required for SCC, (e) has shown that the ID surface of the SG tube will be likely to be in a state of residual compressive stress, and (f) has limited the amount of time that the sleeves will be in service to eight fuel cycles.

The NRC staff finds that the licensee has demonstrated the acceptability of the leak-limiting Alloy 800 sleeve repair in accordance with Appendix B to 10 CFR Part 50; GDC 14, 15, 19, 30, 31, and 32 of Appendix A to 10 CFR Part 50; Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976; and the ASME Code. As noted above, the TZ leak-limiting Alloy 800 sleeves are limited to eight fuel cycles of operation after installation. There is no limit on the number of operating fuel cycles for TS leak-limiting Alloy 800 sleeves.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment on December 27, 2018. The State official had no comments.

5.0 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 (83 FR 26105). Accordingly, the amendments meet 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.

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

Principal Contributor: A. Johnson

Date: February 25, 2019

SUBJECT: BEAVER VALLEY POWER STATION, UNIT 2 – ISSUANCE OF AMENDMENT NO. 193 RE: REVISE STEAM GENERATOR TECHNICAL SPECIFICATIONS (EPID L-2018-LLA-0075) DATED FEBRUARY 25, 2019

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