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

December 16, 2015

Mr. Eric A. Larson, 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 NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS RE: LICENSE AMENDMENT REQUEST TO REVISE STEAM GENERATOR TECHNICAL SPECIFICATIONS (CAC NOS. MF6054 AND 6055)

Dear Mr. Larson:

The Commission has issued the enclosed Amendment No. 296 to Renewed Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1, and Amendment No. 184 to Renewed Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit No. 2. These amendments consist of changes to the Technical Specifications in response to your application dated April 1, 2015, as supplemented by letter dated August 10, 2015.

The amendments revise various sections associated with steam generators and include changes that are consistent with guidance provided in Technical Specification Task Force Traveler-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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,

Taylor A. Lamb

Taylor A. Lamb, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment No. 296 to DPR-66
2. Amendment No. 184 to NPF-73
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION, LLC

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 296
Renewed License No. DPR-66

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 April 1, 2015, as supplemented by letter dated August 10, 2015, 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. DPR-66 is hereby amended to read as follows:

    (2) <u>Technical Specifications</u>

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 296, 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A Broaddus

Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: December 16, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 296

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 a marginal line indicating the area of change.

Remove	Insert
Page 3	Page 3

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

Remove	Insert
3.4.20-1	3.4.20-1
3.4.20-2	3.4.20-2
5.5-4	5.5-4
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.5-11	5.5-11
5.5-12	5.5-12
5.5-13	5.5-13
5.5-14	5.5-14
5.5-15	5.5-15
5.5-16	5.5-16
5.6-4	5.6-4
5.6-5	5.6-5
5.6-6	5.6-6

(3) FENOC, 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) FENOC, 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) FENOC, 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

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. 296, 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)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.20 Steam Generator (SG) Tube Integrity

LCO 3.4.20 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging or repair criteria shall be plugged or repaired[1] in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

- NOTE -

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging or repair criteria and not plugged or repaired[1] in accordance with the Steam Generator Program.	A.1	Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND		
	A.2	Plug or repair[1] the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

[1] SG Tube repair is only applicable to Unit 2.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.20.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.20.2	Verify that each inspected SG tube that satisfies the tube plugging or repair criteria is plugged or repaired[1] in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

[1] SG Tube repair is only applicable to Unit 2.

5.5 Programs and Manuals

5.5.4 Inservice Testing Program (continued)

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies and to other normal and accelerated Frequencies specified as 2 years or less in the Inservice Testing Program for performing inservice testing activities,

c. The provisions of SR 3.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME OM Code shall be construed to supersede the requirements of any TS.

5.5.5 Steam Generator (SG) Program

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

5.5.5.1 Unit 1 SG Program

a. Provisions for Condition Monitoring Assessments

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

b. Provisions for Performance Criteria for SG Tube Integrity

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

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the

5.5 Programs and Manuals

5.5.5.1 Unit 1 SG Program (continued)

design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

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

5.5 Programs and Manuals

5.5.5.1 Unit 1 SG Program (continued)

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 of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program

a. Provisions for Condition Monitoring Assessments

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

b. Provisions for Performance Criteria for SG Tube Integrity

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

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

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

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

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

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

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

c. Provisions for SG Tube Plugging or Repair Criteria

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

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

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

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

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

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

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

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

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

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

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

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

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

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

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

where:

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

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

NDE = 95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC). The NDE is the value provided by the NRC in GL 95-05 as supplemented.

Implementation of these mid-cycle repair limits should follow the same approach as in Specifications 5.5.5.2.c.4.a through 5.5.5.2.c.4.d.

5. The F* methodology, as described below, may be applied to the expanded portion of the tube in the hot-leg or cold-leg tubesheet region as an alternative to the 40% depth based criteria of Specification 5.5.5.2.c.1:

   a) Tubes with no portion of a lower sleeve joint in the hot-leg or cold-leg tubesheet region shall be repaired or plugged upon detection of any flaw identified within 3.0 inches below the top of the tubesheet or within 2.22 inches below the bottom of roll transition, whichever elevation is lower. Flaws located below this elevation may remain in service regardless of size.

   b) Tubes which have any portion of a sleeve joint in the hot-leg or cold-leg tubesheet region shall be plugged upon detection of any flaw identified within 3.0 inches below the lower end of the lower sleeve joint. Flaws located greater than 3.0 inches below the lower end of the lower sleeve joint may remain in service regardless of size.

   c) The F* methodology cannot be applied to the tubesheet region where a laser or TIG welded sleeve has been installed.

d. Provisions for SG Tube Inspections

-NOTE-

The requirement for methods of inspection 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 does not apply to the portion of the original tube wall adjacent to the nickel band (the lower half) of the lower joint for the repair process that is discussed in Specification 5.5.5.2.f.3. However, the method of inspection in this area shall be a rotating plus point (or equivalent) coil. The SG tube plugging criterion of Specification 5.5.5.2.c.3 is applicable to flaws in this area.

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

5.5 Programs and Manuals

5.5.5.2 Unit 2 SG Program (continued)

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 or repair criteria. The tube-to-tubesheet weld is not part of the tube. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. In addition to meeting the requirements of d.1, d.2, d.3, d.4 and d.5 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 steam generator at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage. 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 of degradation at this location and that may satisfy the applicable tube plugging or repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.

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 (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.

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 shall remain in service for no more than five fuel cycles of operation starting from the outage when the sleeve was installed.

5.5 Programs and Manuals

5.5.6 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off control point chemistry conditions, and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.7 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems for the Control Room Emergency Ventilation System (CREVS) and the Supplemental Leak Collection and Release System (SLCRS).

Tests described in Specifications 5.5.7.a and 5.5.7.b shall be performed at least once per 18 months and after the following:

- Each complete or partial replacement of the high efficiency particulate air (HEPA) filter or charcoal adsorber bank;
- Any structural maintenance on the HEPA filter or charcoal adsorber housing;
- Significant painting, fire, or chemical release (for the Unit 1 and Unit 2 SLCRS) in any ventilation zone communicating with the system while the filtration system is operating; and
- Significant painting, fire, or chemical release (for the Unit 1 and Unit 2 CREVS) in the vicinity of control room outside air intakes while the system is operating.

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

Tests described in Specification 5.5.7.c shall be performed at least once per 18 months and after the following:

- 720 hours of adsorber operation (for the Unit 1 and 2 CREVS and the Unit 1 SLCRS) or after 4 months of adsorber operation (for the Unit 2 SLCRS);
- Any structural maintenance on the charcoal adsorber bank housing;
- Significant painting, fire, or chemical release (for the Unit 1 and Unit 2 SLCRS) in any ventilation zone communicating with the system while the filtration system is operating; and
- Significant painting, fire, or chemical release (for the Unit 1 and Unit 2 CREVS) in the vicinity of control room outside air intakes while the system is operating.

Tests described in Specifications 5.5.7.d and 5.5.7.e shall be performed at least once per 18 months.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

a. Demonstrate for each of the required ESF systems that an inplace test of the HEPA filters shows a penetration and system bypass specified below when tested in accordance with ANSI N510-1980 (for the Unit 1 and 2 CREVS) and the Unit 2 SLCRS and in accordance with ANSI N510-1975 (for the Unit 1 SLCRS) at the system flowrate specified below:

ESF Ventilation System	Penetration	Flowrate
SLCRS	$<$ 1.0% (Unit 1) $<$ 0.05% (Unit 2)	$\geq$ 32,400 cfm and $\leq$ 39,600 cfm (Unit 1) $\geq$ 51,300 cfm and $\leq$ 62,700 cfm (Unit 2)
CREVS	$<$ 0.05%	$\geq$ 800 cfm and $\leq$ 1000 cfm

b. Demonstrate for each of the required ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass specified below when tested in accordance with ANSI N510-1980 (for the Unit 1 and 2 CREVS and the Unit 2 SLCRS) and ANSI N510-1975 (for the Unit 1 SLCRS) at the system flowrate specified below:

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Penetration	Flowrate
SLCRS	< 1.0% (Unit 1) < 0.05% (Unit 2)	≥ 32,400 cfm and ≤ 39,600 cfm (Unit 1) ≥ 51,300 cfm and ≤ 62,700 cfm (Unit 2)
CREVS	< 0.05%	≥ 800 cfm and ≤ 1000 cfm

c. Demonstrate for each of the required ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, or using a slotted tube sampler in accordance with ANSI N509-1980 shows, within 31 days after removal, the methyl iodide removal efficiency greater than or equal to the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C, an inlet methyl iodide concentration of 1.75 mg/m^3, and an air flow velocity and relative humidity (RH) specified below:

ESF Ventilation System	Removal Efficiency	Air Flow Velocity	RH
SLCRS	90% (Unit 1) 99% (Unit 2)	0.9 ft/sec (Unit 1) 0.7 ft/sec (Unit 2)	≥ 95% (Unit 1) ≥ 70% (Unit 2)
CREVS	99% (Unit 1) 99% (Unit 2)	0.68 ft/sec (Unit 1) 0.7 ft/sec (Unit 2)	≥ 70% (Unit 1) ≥ 70% (Unit 2)

d. Demonstrate for each of the required ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified as follows:

ESF Ventilation System	Delta P	Flowrate
SLCRS	6 inches Water Gauge (Unit 1) 6.8 inches Water Gauge (Unit 2)	≥ 32,400 cfm and ≤ 39,600 cfm (Unit 1) ≥ 51,300 cfm and ≤ 62,700 cfm (Unit 2)
CREVS	6 inches Water Gauge (Unit 1) 5.6 inches Water Gauge (Unit 2)	≥ 800 cfm and ≤ 1000 cfm (Unit 1) ≥ 800 cfm and ≤ 1000 cfm (Unit 2)

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1980.

ESF Ventilation System	Wattage
SLCRS	≥ 160.9 kW and ≤ 264.5 kW (Unit 2 only)
CREVS	≥ 3.87 kW and ≤ 5.50 kW

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in waste gas decay tanks (Unit 1) and gaseous waste storage tanks (Unit 2), and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall ensure that the concentration of hydrogen and oxygen is maintained below flammability limits,

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank (Unit 1) and each connected group of waste gas storage tanks (Unit 2) is less than the amount that would result in a whole body exposure of > 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents, and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations greater than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.6 Reporting Requirements

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

WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

The methodology listed in WCAP-14040-NP-A was used with two exceptions:

- ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1."
- ASME, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," 1996 version.

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

5.6.5 Post Accident Monitoring Report

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

5.6.6 Steam Generator (SG) Tube Inspection Report

5.6.6.1 Unit 1 SG Tube Inspection Report

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

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service-induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

5.6 Reporting Requirements

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

5.6.6.2 Unit 2 SG Tube Inspection Report

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

   a. The scope of inspections performed on each SG,

   b. Degradation mechanisms found,

   c. Nondestructive examination techniques utilized for each degradation mechanism,

   d. Location, orientation (if linear), and measured sizes (if available) of service-induced indications,

   e. Number of tubes plugged or repaired during the inspection outage for each degradation mechanism,

   f. The number and percentage of tubes plugged or repaired to date, and the effective plugging percentage in each steam generator,

   g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

   h. Repair method utilized and the number of tubes repaired by each repair method.

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

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

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

5.6 Reporting Requirements

5.6.6.2 Unit 2 SG Tube Inspection Report (continued)

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

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

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

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

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

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

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION, LLC

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 184
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 April 1, 2015, as supplemented by letter dated August 10, 2015, 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.

Enclosure 2

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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 184, 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 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Douglas A Broaddus

Douglas A. Broaddus, Chief
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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

Date of Issuance: December 16, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 184

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 lines indicating the area of change.

Remove	Insert
Page 4	Page 4

Beaver Valley Power Station Units 1 and 2 share a common Appendix A, Technical Specifications. As such, the replacement pages listed in the attachment to License Amendment No. 296 will also be applicable for Amendment No. 184.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the term or conditions of any lease agreements executed as part of these transactions; (ii) the BVPS Operating Agreement, (iii) the existing property insurance coverage for BVPS Unit 2, and (iv) any action by a lessor or others that may have adverse effect on the safe 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 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. 184, 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 296 AND 184 TO RENEWED

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRST ENERGY NUCLEAR GENERATION, LLC

BEAVER VALLEY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-334 AND 50-412

1.0 INTRODUCTION

By application dated April 1, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15092A569), as supplemented by letter dated August 10, 2015 (ADAMS Accession No. ML15222B113), FirstEnergy Nuclear Operating Company, et al. (the licensee), requested changes to the Technical Specifications (TSs) for Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2).

The licensee proposed changes that would revise various sections of the TSs associated with steam generators, including changes consistent with guidance provided in Technical Specification Task Force Traveler (TSTF)-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection" (ADAMS Accession No. ML110610350).

The supplement dated August 10, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on May 12, 2015 (80 FR 27198).

2.0 REGULATORY EVALUATION

The licensee requested a change to the Facility Operating Licensee for BVPS-1 and BVPS-2, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90, "Application for amendment of license, construction permit, or early site permit." The regulations in Appendix A 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 General Design Criterion (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."

Both BVPS-1 and BVPS-2 were evaluated against the GDC requirements discussed above. Although the BVPS-1 construction permit was issued in June of 1970, prior to the GDC being published as Appendix A to 10 CFR in July of 1971, Appendix A1 of the BVPS-1 UFSAR notes that the design conforms with the intent of GDC 14, 15, 30, 31, and 32. The BVPS-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, "Codes and standards," components which are part of the RCPB must meet the requirements of Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (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, "Rules for Inservice Inspection of Nuclear Power Plant Components," 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, "Reactor Site Criteria," guidelines for offsite doses (or 10 CFR 50.67, "Accident source term," as appropriate); GDC 19, "Control Room," criteria for control room operator doses or some fraction thereof as appropriate to the accident; or the U.S. Nuclear Regulatory Commission (NRC)-approved licensing basis.

In Section 50.36 of 10 CFR, "Technical specifications," 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 (SRs); (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in the plant's TSs.

The LCOs (and accompanying action statements) and the SRs in the Standard Technical Specifications (STS) that are relevant to SG tube integrity are in Specification 3.4.13, "Reactor Coolant System (RCS) Operational Leakage," and Specification 3.4.20 (SR 3.4.20.2), "Steam Generator (SG) Tube Integrity." The SRs in the "Steam Generator (SG) Tube Integrity" specification reference the SG Program, which is defined in the STS administrative controls. BVPS-1 and BVPS-2, TSs 3.4.13 and 3.4.20, address requirements similar to those specified in the STS sections above.

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, while the reporting requirements relating to implementation of the SG Program are defined in TS 5.6.6.

TS 5.5.5, "Steam Generator (SG) Program," for BVPS-1 and BVPS-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.a requires a condition monitoring assessment be performed during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met. TS 5.5.5.d includes provisions regarding the scope, frequency, and methods of SG tube inspections.

3.0 TECHNICAL EVALUATION

The SGs at BVPS-1 were replaced in 2006 with three Westinghouse Model 54F SGs. Each SG contains 3,592 thermally treated Alloy 690 tubes. The three SGs at BVPS-2 are Westinghouse Model 51 SGs. Each SG contains 3,388 mill-annealed Alloy 600 tubes. The SGs at BVPS-2 are scheduled to be replaced in 2020.

3.1 Description of Proposed Changes

The proposed changes can be separated into three categories. The first category of changes concerns those changes proposed by BVPS that are consistent with Revision 2 to TSTF-510. The proposed changes revise TS 3.4.20, "Steam Generator (SG) Tube Integrity"; TS 5.5.5, "Steam Generator (SG) Program"; and TS 5.6.6, "Steam Generator Tube Inspection Report."

The second category of changes involves those changes in the submittal that are inconsistent with TSTF-510, Revision 2, and the model application. Since these changes are inconsistent with TSTF-510, Revision 2, and are not approved as part of the model safety evaluation (SE), the NRC staff reviewed these changes separately from those changes that were consistent with TSTF-510, Revision 2, mentioned above.

The third category of changes concerns BVPS-2 TS 5.5.5.2.f, "Provisions for SG Tube Repair Methods." TS 5.5.5.2.f currently allows the licensee to use Alloy 800 sleeves as a method to repair SG tubes until 2017, irrespective of the date of installation of the sleeve. The proposed change would allow the licensee to use Alloy 800 sleeves as a tube repair method for up to 7 1/2 years (five fuel cycles of operation) after the sleeve installation.

3.2 TSTF-449 and TSTF-510 Background

The SG tubes of a PWR have a number of important safety functions. They are an integral part of the RCPB and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon to transfer heat from the primary to secondary system. In addition, SG tube integrity is relied upon to prevent uncontrolled fission product release under conditions resulting from core damage during severe accidents.

The current SG TS requirements for BVPS were based on TSTF-449, Revision 4, "Steam Generator Tube Integrity" (NRC May 6, 2015, *Federal Register* Notice of Availability (70 FR 24126)). The TSTF-449 changes to the STS incorporated a new, largely performance-based approach for ensuring the integrity of the tubes is maintained. The performance-based requirements were supplemented by prescriptive requirements relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and corrected on a timely basis. In September 2006, the TSTF-449, Revision 4, changes were adopted in the plant TSs by BVPS-1 and BVPS-2. The changes included in TSTF-510 addressed implementation issues associated with the inspection periods and addressed other administrative changes and clarifications.

3.3 Previously Implemented Changes Consistent with TSTF-510, Revision 2

The changes in TSTF-510, Revision 2, reflect licensees' early implementation experience with TSs based on TSTF-449. The changes in TSTF-510, Revision 2, are editorial corrections, changes, and clarifications intended to improve internal consistency and consistency with implementing industry documents and usability without changing the intent of the requirements. The proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections. The NRC staff approved TSTF-510, Revision 2, for use with the consolidated line item improvement process on October 19, 2011 (ADAMS Accession No. ML112101604). The staff's model SE is available under ADAMS Accession No. ML112101513. Other than the variations or deviations discussed in Section 3.4 below, the licensee is not proposing any variations from the TS changes described in TSTF-510, Revision 2.

The staff notes that the following TS sections were already consistent with TSTF-510, Revision 2, prior to the submittal of the license amendment request (LAR). Thus, these TSs required no change to make them consistent with TSTF-510, Revision 2. The portions that were already consistent with TSTF-510, Revision 2, are underlined.

- 5.5.5.1.d – 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.

- 5.5.5.2.d – 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.

3.4 Changes Inconsistent with TSTF-510, Revision 2

The BVPS-1 and BVPS-2 TSs utilize different numbering than the STS on which TSTF-510, Revision 2, was based.

For BVPS-1 and BVPS-2, the "Steam Generator (SG) Program" in the TS is numbered 5.5.5 rather than 5.5.9, and the "Steam Generator Tube Inspection Report" is numbered 5.6.6 rather than 5.6.7.

These differences are administrative and do not affect the applicability of TSTF-510, Revision 2, to the BVPS-1 and BVPS-2 TSs; therefore, the staff finds these differences acceptable.

The following TSs contain wording changes that are inconsistent with TSTF-510, Revision 2. Inconsistent additions and deletions proposed by the licensee are noted with an **underline in bold** and ~~**strikethrough in bold**~~, respectively.

TS	**Licensee Proposed**	**TSTF-510**
5.5.5.1	Unit 1 ~~**Steam Generator**~~SG Program	Steam Generator (SG) Program
5.5.5.1.b	**Provisions for** Performance Criteria for SG Tube Integrity	Performance criteria for SG tube integrity
5.5.5.2	Unit 2 ~~**Steam Generator**~~ SG Program	Steam Generator (SG) Program
5.5.5.2.b	**Provisions for** Performance Criteria for SG Tube Integrity	Performance criteria for SG tube integrity
5.5.5.2.b.3	The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational ~~**Leakage**~~**LEAKAGE**."	The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."
5.6.6	Steam Generator **(SG)** Tube Inspection Report	Steam Generator Tube Inspection Report
5.6.6.1	**Unit 1 SG Tube Inspection Report** A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.1, "Unit 1 ~~**Steam Generator (**~~SG~~**)**~~ Program." The report shall include:	A report shall be submitted within 180 days after initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:
5.6.6.1.d	Location, orientation (if linear), and measured sizes (if available) of service-induced indications	Location, orientation (if linear), and measured sizes (if available) of service induced indications

5.6.6.1.f	The number and percentage of tubes plugged or repaired to date, and the effective plugging percentage in each steam generator, **<u>and</u>**	The number and percentage of tubes plugged or repaired to date, and the effective plugging percentage in each steam generator,
5.6.6.2	**<u>Unit 2 SG Tube Inspection Report</u>** **<u>1.</u>** A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.2, "**<u>Unit 2</u>** **~~Steam Generator (~~**SG**~~)~~** Program." The report shall include:	A report shall be submitted within 180 days after initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, "Steam Generator (SG) Program." The report shall include:
5.6.6.2.1.d	Location, orientation (if linear), and measured sizes (if available) of service**<u>-</u>**induced indications	Location, orientation (if linear), and measured sizes (if available) of service induced indications
5.6.6.2.1.g	The results of condition monitoring, including the results of tube pulls and in-situ testing, **<u>and</u>**	The results of condition monitoring, including the results of tube pulls and in-situ testing,
5.6.6.2.2	A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.5.2, "Unit 2 **~~Steam Generator~~****<u>SG</u>** Program," when voltage-based alternate repair criteria have been applied.	A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, "Steam Generator (SG) Program."

The above deviations from TSTF-510, Revision 2, are administrative in nature and do not affect the regulatory or technical evaluations of the model SE. The staff has determined the above deviations to be acceptable.

The licensee proposed the following changes to the alternate repair criteria specific to BVPS-2. The purpose of the changes was to clarify that the tubes could be either plugged or repaired (as opposed to just repaired). Additions and deletions proposed by the licensee are noted with an **<u>underline in bold</u>** and **~~strikethrough in bold~~**, respectively.

<u>TS</u>	**<u>Licensee Proposed</u>**
5.5.5.2.c.1	Tubes found by inservice inspection to contain a flaw in a non-sleeved region with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired except if permitted to remain in service through

	application of the alternate **plugging or** repair criteria discussed in Specification 5.5.5.2.c.4 or 5.5.5.2.c.5.
5.5.5.2.c.4	Tube support plate voltage-based **plugging or** repair criteria may be applied as an alternative to the 40% depth based criteria of Specification 5.5.5.2.c.1. Tube Support Plate Plugging Limit is used for the disposition of an Alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging **or** ~~(~~repair~~)~~ limit is described below:
5.5.5.2.c.4.b	Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be ~~**repaired or**~~ plugged **or repaired**, except as noted in 5.5.5.2.c.4.c below.
5.5.5.2.d.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.
5.5.5.2.d.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).
5.6.6.2.3	For implementation of the voltage-based **plugging or** repair criteria to tube support plate intersections, notify the Commission prior to returning the steam generators to service (MODE 4) should any of the following conditions arise:

The above deviations from TSTF-510, Revision 2, are administrative in nature and do not affect the regulatory or technical evaluations of the LAR or the approved alternate repair criteria. The staff has determined the above deviations to be acceptable.

3.5 TS 5.5.5.2.f Background

By application dated October 10, 2008 (ADAMS Accession No. ML082890823), the licensee requested changes to TS 5.5.5 for BVPS-2. The proposed changes revised TS 5.5.5 to allow the use of Westinghouse leak-limiting Alloy 800 SG tube sleeves as a repair method for SG tubes. By letter dated September 30, 2009 (ADAMS Accession No. ML092590189), the NRC staff approved an LAR for the use of Alloy 800 sleeves as an acceptable repair method at BVPS-2, with the provision that all sleeves be removed by the spring of 2017. The SGs at BVPS-2 were originally scheduled for replacement in the spring of 2017. The licensee evaluated the service life of the sleeves to meet this replacement date; however, the first use of Alloy 800 sleeves occurred in the fall of 2012, and the steam generator replacement date has been delayed until the spring of 2020.

3.6 Changes to TS 5.5.5.2.f, "Provisions for SG Tube Repair Methods"

The licensee proposed the following revision to the TS Section 5.5.5.2 for BVPS-2 SG Program. Additions are noted in **bold** and deletions with a ~~strikethrough~~.

5.5.5.2.f. Provisions for SG Tube Repair Methods

3. Westinghouse leak-limiting Alloy 800 sleeves, WCAP-15919-P, Revision 2. ~~All Alloy 800 sleeves shall be removed from service by the spring of 2017 Unit 2 refueling outage (2R19).~~ **An Alloy 800 sleeve shall remain in service for no more than five fuel cycles of operation starting from the outage when the sleeve was installed.**

The licensee has proposed to remove the requirement that all Westinghouse leak-limiting Alloy 800 sleeves shall be removed by the spring of 2017 from the BVPS-2 SGs. A sleeve is a tube segment that is inserted into a tube in an existing SG and expanded at both ends of the sleeve to form a structural joint.

On September 30, 2009 (ADAMS Accession No. ML092590189), the NRC staff approved an LAR for the leak-limiting Alloy 800 sleeve design for BVPS-2. The sleeve design for BVPS-2 contains a nickel band, behind which there is limited inspection capability of the parent tube. As a result of the limited inspection capability, a limit on the amount of time that the sleeves were to be in service was adopted. Since the amendment was approved in the fall of 2009, the 2017 date would result in the sleeves being in service for no more than five operating cycles. This time limit of five operating cycles on the service life of a sleeve was intended to limit the extent of sleeve joint degradation should it occur. The SE for the September 30, 2009, amendment states:

> Although there is only limited data demonstrating the capability to reliably detect flaws in the parent tube behind the nickel band region of the Alloy 800 sleeve, the NRC staff finds the licensee's inspection program acceptable, since (a) the licensee 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) the licensee has demonstrated that severe degradation in the joints can be detected, (c) the licensee has determined that the axial load carrying capability of the joint is not compromised in the even that severe degradation is present behind the nickel band region of the Alloy 800 sleeve, and (d) the licensee has limited the amount of time that the sleeves will be in service by proposing a TS requirement to remove all Alloy 800 sleeves from service by the spring of 2017 BVPS-2 refueling outage (2R19). The limitation on the service life of the sleeve limits the amount of time that degradation of the sleeve joint could occur.

Furthermore, the September 30, 2009, SE states:

> Westinghouse's structural analysis also included calculations for a minimum required sleeve thickness based on ASME Code, Section III. The calculations show that the actual sleeve wall thickness is greater than the minimum required thickness, and, therefore, is structurally acceptable. Westinghouse also calculated the percentage of sleeve wall thickness that could be degraded. This calculation considered axial and circumferential cracking. The calculated amount of degradation that could be tolerated and still met 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.

Alloy 800 sleeves were first used at BVPS-2 in the fall of 2012. In light of delay in the Alloy 800 sleeves' first use, the licensee is proposing to change TS 5.5.5.2.f as described above. In the staff's previous evaluation of Alloy 800 sleeves, the staff concluded that Alloy 800 sleeves could be used for five operation cycles. Upon review, the staff has determined that the conclusion remains valid. Thus, the staff finds this change to be acceptable, since it limits the service life of the sleeve to five fuel cycles.

3.7 Technical Evaluation Conclusion

The NRC staff has reviewed the licensee's submittal to revise the SG TSs and concludes that the proposed changes will not reduce the levels of assurance of SG tube integrity as compared to the current TSs. The changes consistent with TSTF-510 have been approved by the staff as part of the consolidated line item improvement process and are found to be acceptable. The changes listed in Section 3.4 that are inconsistent with TSTF-510 are also found to be acceptable, since they are administrative in nature and do not change any technical requirements. The changes to TS 5.5.5.2.f associated with the Alloy 800 leak-limiting sleeves are found to be acceptable, since the sleeves were previously credited with a service life of up to five cycles of operation. The licensee's request maintains the requirement that an Alloy 800 sleeve shall remain in service for no more than five cycles of operation.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendment. 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

(80 FR 27198). 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: Alan Huynh

Dated: December 16, 2015

December 16, 2015

Mr. Eric A. Larson, 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 NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS RE: LICENSE AMENDMENT REQUEST TO REVISE STEAM GENERATOR TECHNICAL SPECIFICATIONS (CAC NOS. MF6054 AND 6055)

Dear Mr. Larson:

The Commission has issued the enclosed Amendment No. 296 to Renewed Facility Operating License No. DPR-66 for the Beaver Valley Power Station, Unit No. 1, and Amendment No. 184 to Renewed Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit No. 2. These amendments consist of changes to the Technical Specifications in response to your application dated April 1, 2015, as supplemented by letter dated August 10, 2015.

The amendments revise various sections associated with steam generators and include changes that are consistent with guidance provided in Technical Specification Task Force Traveler-510, Revision 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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,
/RA/
Taylor A. Lamb, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-334 and 50-412

Enclosures:
1. Amendment No. 296 to DPR-66
2. Amendment No. 184 to NPF-73
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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ADAMS Accession No.: ML15294A439 *via memo

OFFICE	DORL/LPLI-2/PM	DORL/LPLI-2/LA	NRR/DE/ESGB*	NRR/DSS/STSB	OGC – NLO
NAME	TLamb	LRonewicz	GKulesa	RElliott	BHarris (w/edits)
DATE	11/2/2015	12/7/2015	9/24/2015	11/4/2015	12/3/2015
OFFICE	DORL/LPL1-2/BC	DORL/LPL1-2/PM			
NAME	DBroaddus	TLamb			
DATE	12/15/2015	12/16/2015			

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