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

September 30, 2009

Mr. Peter P. Sena III
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 NO. 2 - ISSUANCE OF AMENDMENT RE: THE USE OF WESTINGHOUSE LEAK-LIMITING ALLOY 800 SLEEVES FOR STEAM GENERATOR TUBES REPAIR (TAC NO. MD9969)

Dear Mr. Sena:

The Commission has issued the enclosed Amendment No. 170 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 10, 2008, as supplemented by letters dated June 16 and July 14, 2009.

The amendment revises TS 5.5.5 to allow an additional method of repair for steam generator (SG) tubes by installation of leak limiting Alloy 800 sleeves developed by Westinghouse and clarifies an existing reporting requirement in TS 5.6.6.2.4 concerning SG tube inspections.

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

Sincerely,

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosures:
1. Amendment No. 170 to NPF-73
2. Safety Evaluation

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

DOCKET NO. 50-412

BEAVER VALLEY POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 170
License No. NPF-73

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

   A. The application for amendment by FirstEnergy Nuclear Operating Company, et al. (FENOC, licensee), dated October 10, 2008, as supplemented by letters dated June 16 and July 14, 2009, 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.

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 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.170, 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 prior to achieving Mode 4 during startup from the fall 2009 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

John P. Boska for

Nancy L. Salgado, Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the License and
Technical Specifications

Date of Issuance: September 30, 2009

ATTACHMENT TO LICENSE AMENDMENT NO. 170

FACILITY OPERATING LICENSE NO. NPF-73

DOCKET NO. 50-412

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

Remove Page	Insert Page
3a	3a

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

Remove Pages	Insert Pages
5.5.8	5.5.8
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
5.5-17	5.5-17
5.5-18	5.5-18
5.5-19	5.5-19
5.5-20	5.5-20
5.5-21	5.5-21
5.6-6	5.6-6

transactions shall have no effect on the license for the BVPS Unit 2 facility throughout the term of the license.

(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 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. 170, 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 Steam Generator (SG) Program (continued)

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 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 (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 repaired or plugged, except as noted in 5.5.5.2.c.4.c below.

   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.

5.5 Programs and Manuals

5.5.5.2 Unit 2 Steam Generator (SG) Program (continued)

5. The F* methodology, as described below, may be applied to the expanded portion of the tube in the hot-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 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 2.2 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 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.

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 repair 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 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 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, d.5 and d.6 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.

5.5 Programs and Manuals

5.5.5.2 Unit 2 Steam Generator (SG) Program (continued)

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

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one interval between refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one interval between refueling outages (whichever is less). 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.

4. Indications left in service as a result of application of the tube support plate voltage-based 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 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.

5. 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 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).

6. 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

5.5 Programs and Manuals

5.5.5.2 Unit 2 Steam Generator (SG) Program (continued)

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. All Alloy 800 sleeves shall be removed from service by the spring of 2017 Unit 2 refueling outage (2R19).

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 Programs and Manuals

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.

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.

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

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)	≥ 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

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:

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:

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

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)

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 Programs and Manuals

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.

5.5 Programs and Manuals

5.5.9 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. An API gravity or an absolute specific gravity within limits,

   2. A flash point and kinematic viscosity (if gravity was not determined by comparison with suppliers certification) within limits for ASTM 2D fuel oil, and

   3. A water and sediment content within limits.

b. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil, and

c. Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the license or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

5.5 Programs and Manuals

5.5.10 Technical Specifications (TS) Bases Control Program (continued)

d. Proposed changes that meet the criteria of Specification 5.5.10.b.1 and 5.5.10.b.2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.11 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

a. The SFDP shall contain the following:

   1. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists,

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities, and

   4. Other appropriate limitations and remedial or compensatory actions.

b. A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

   1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable, or

   2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

   3. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Containment Leakage Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. For Unit 1, exemptions to Appendix J of 10 CFR 50 are dated November 19, 1984, December 5, 1984, and July 26, 1995. For Unit 2, exemptions to Appendix J of 10 CFR 50 are as stated in the Operating License. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September, 1995, as modified by the following exceptions:

   1. For Unit 1, the next Type A test performed after the May 29, 1993 Type A test shall be performed no later than May 28, 2008.

   2. For Unit 2, the next Type A test performed after the November 10, 1993 Type A test shall be performed no later than November 9, 2008.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 43.1 psig (for Unit 1) and 44.8 psig (for Unit 2).

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup prior to MODE 4 entry following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

   2. Air lock testing acceptance criteria are:

      a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

5.5 Programs and Manuals

5.5.12 Containment Leakage Rate Testing Program (continued)

b) For each emergency air lock door, no detectable seal leakage when gap between door seals is pressurized to ≥ 10 psig or door seal leakage quantified to ensure emergency air lock door seal leakage rate is ≤ 0.0005 L_a when tested at ≥ 10 psig.

c) For each personnel air lock door, no detectable seal leakage when gap between door seals is pressurized to ≥ P_a or door seal leakage quantified to ensure personnel air lock door seal leakage rate is ≤ 0.0005 L_a when tested at ≥ P_a.

e. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

f. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5.13 Battery Monitoring and Maintenance Program

This Program provides for battery restoration and maintenance, which includes the following:

a. Actions to restore battery cells with float voltage < 2.13 V,

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the top of the plates, and

c. Actions to verify the remaining cells are ≥ 2.07 V when a cell or cells have been found to be < 2.13 V.

5.5.14 Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident. The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

5.5 Programs and Manuals

5.5.14 Control Room Envelope Habitability Program (continued)

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREVS, operating at the flow rate required by the VFTP, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the periodic assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

5.6 Reporting Requirements

5.6.6.2 Unit 2 Steam Generator 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. The report shall include the following hot-leg tubesheet region inspection results associated with the application of F*:

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 170 TO FACILITY OPERATING LICENSE NO. NPF-73

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

OHIO EDISON COMPANY

THE TOLEDO EDISON COMPANY

BEAVER VALLEY POWER STATION, UNIT NO. 2

DOCKET NO. 50-412

1.0 INTRODUCTION

By application dated October 10, 2008, as supplemented by letters dated June 16 and July 14, 2009, the FirstEnergy Nuclear Operating Company (licensee), requested changes to the Technical Specifications (TSs) for Beaver Valley Power Station, Unit No. 2 (BVPS-2). The supplements dated June 16 and July 14, 2009, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 17, 2009 (74 FR 7482).

The proposed changes would revise TS 5.5.5 to allow the use of Westinghouse leak-limiting Alloy 800 steam generator (SG) tube sleeves as an additional repair method for SG tubes. The proposed changes would also clarify an existing reporting requirement in TS 5.6.6.2.4 concerning SG tube inspections.

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements and Guidance

The applicable NRC regulations and guidance for review of the proposed sleeve repair method are as follows:

General Design Criterion 14 of Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, requires that the reactor coolant pressure boundary (RCPB) be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. SG tubes represent a significant part of the

RCPB. When a flaw size exceeds the repair criteria, the current TSs require that the tube be either repaired using a sleeve or removed from service by plugging. To repair a part of the existing RCPB, 10 CFR 50.55a requires that the sleeve repair method be qualified in accordance with Section XI of the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), which refers to Section III of the ASME Code that is part of the design basis for the SG tubing. The sleeve must satisfy all applicable ASME Code Section III limits for design, operating conditions, and accident loading conditions. In addition, the sleeve wall thickness needs to satisfy the minimum wall thickness requirement of the ASME Code, and the structural and leakage integrity requirements in the plant's TSs.

Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires a quality assurance program for the design, fabrication, construction, and operation of structures, systems, and components in nuclear plants. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those structures, systems, and components. These activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying.

Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [Pressurized-Water Reactor] Steam Generator Tubes," provides guidance for determining the minimum SG tube wall thickness and for determining the repair criteria for SG tubes with sleeves. In accordance with RG 1.121, the margin of safety against tube rupture under normal operating conditions should not be less than three at any tube location where flaws have been detected. The margin of safety against tube failure under postulated accidents, such as a loss-of-coolant accident (LOCA), main steam line break, or feedwater line break concurrent with the safe shutdown earthquake, should be consistent with the margin of safety determined by the stress limits specified in Section III of the ASME Code.

2.2 Proposed TS Changes

The licensee proposed the following revisions to TS Section 5.5.5.2 Unit 2 SG Program. Additions are noted in **bold;** deletions with a ~~strikethrough~~.

5.5.5.2.c. Provisions for SG Repair Criteria

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**

5.5.5.2.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 repair 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 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 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, **d.5** and d.**6**~~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.

[d.1 through d.5 not shown]

6. **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

[f.1. and f.2 not shown]

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).**

5.6.6.2 Unit 2 SG Tube Inspection Report

[2.1 through 2.3 not shown]

4. ~~Report the following information to the NRC~~ **A report shall be submitted** within 90 days after ~~achieving~~ **the initial entry into** MODE 4 following an outage in which the F* methodology was applied. **The report shall include the following hot-leg tubesheet region inspection results associated with the application of F*:**

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

3.0 TECHNICAL EVALUATION

The SGs at BVPS-2 are Westinghouse model 51 SGs. Each SG contains 3,388 mill annealed Alloy 600 tubes. Each tube has a nominal outside diameter (OD) of 0.875-inch and a nominal wall thickness of 0.050-inch. The tubes are supported by a number of carbon steel tube support plates and Alloy 600 anti-vibration bars. The tubes were roll expanded at both ends for the full length of the tubesheet. The entire length of tube within the tubesheet was shot-peened on both the hot- 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 licensee has proposed a tube repair method that uses Westinghouse leak-limiting sleeves made of Alloy 800 material. 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 required to be leak tight. The design, installation, analysis, and qualification tests of the sleeve are documented in the Westinghouse Topical Report, "Steam Generator Tube Repair for Westinghouse Designed Plants with 7/8 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," WCAP-15919-P, Revision 2, dated January 2006 (proprietary). The nonproprietary version of the report is WCAP-15919-NP, Revision 2, dated January 2006. Leak-limiting Alloy 800 sleeves have been approved for use in the original SGs at Calvert Cliffs Nuclear Power Plant Units 1 and 2, Watts Bar Nuclear Plant Unit 1, St. Lucie Plant Unit 2, and Comanche Peak Steam Electric Station Unit 1. Additionally, thousands of these Alloy 800 sleeves have been installed in several international nuclear plants.

Since the NRC staff had previously approved a similar sleeve design, in some cases for only one or two-cycles of operation, the NRC staff focused its review of this submittal on the inspection of the parent tube behind the nickel band and the potential for sleeve collapse, which are the issues that have arisen subsequent to the original approval of this sleeve design. The NRC staff also reviewed the proposed changes to BVPS-2 TS 5.5.5 and TS 5.6.6 to determine the acceptability of the proposed changes.

3.1 Sleeve Design

Two leak-limiting Alloy 800 sleeve designs were proposed for use in repairing a tube. A transition zone (TZ) sleeve is designed to repair tube degradation in the vicinity of the top of the tubesheet. The bottom end of the TZ sleeve, which is the end that is expanded into the tubesheet, includes both a nickel band and a thermally sprayed nickel alloy band. The function of the nickel band is to improve the sealing of the joint, while the function of the thermally sprayed nickel alloy band is to increase the strength of the joint. A tube support sleeve is designed to repair tube degradation at tube support plate intersections or in the freespan region. The lengths of the TZ and tube support sleeves are sized according to the flawed tubing regions which they are designed to repair.

3.2 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.

Prior to sleeve installation, the inside surface of a candidate tube is mechanically conditioned with a high-speed buffing tool. Buffing prepares the tube sealing surface by removing raised material and some of the surface layer oxidation. It was noted that the buffing process may be eliminated in the future when a sufficient confidence level has been developed.

After buffing, 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/tube joints undergo an initial acceptance and baseline inspection using an eddy current technique.

3.3 Sleeve Materials Selection

The sleeve material, Alloy 800, is a nickel-iron-chromium alloy. Westinghouse selected Alloy 800 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. Westinghouse has additional content restriction on various chemical elements and specifies an annealing temperature and yield strength for the Alloy 800 sleeve. This material is acceptable, since it is allowed by the ASME Code, which the NRC staff has approved.

3.4 Sleeve Qualification Testing

Westinghouse performed qualification tests in accordance with Appendix B 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.

3.4.1 Mechanical Testing

Westinghouse performed axial load tests to determine the structural integrity of the sleeve/tube joint. Axial loads are imposed as a result of the differential thermal expansion of the leak-limiting Alloy 800 sleeve and the Alloy 600 tube and the differential pressure across the tube wall. 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 if the parent tube is severed and would not result in tube-to-tube contact in the U-bend area.

Westinghouse performed collapse tests to show that the sleeve would not collapse following a 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 gets high enough in the gap between the tube and the sleeve, the pressure will vent through the joint. In response to an NRC staff request for additional information regarding the range of testing conditions used in the collapse testing, the licensee added a formal regulatory commitment to submit a report to the NRC within 45 days in the unlikely event that a sleeve is found to have inwardly deformed. This report would include a root cause analysis, an assessment of the integrity of the sleeve/parent tube complex and identification of the associated corrective actions.

Westinghouse performed load-cycling tests 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 heat-up 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.

Westinghouse performed leak-rate tests 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 the operational primary-to-secondary leakage limit in the plant TSs and the allowable leakage under accident conditions.

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.

3.4.2 Corrosion Testing

Sleeve/tube joints increase the residual stresses in the parent tube which, in turn, may cause the original tube wall to be susceptible to stress-corrosion cracking (SCC). Westinghouse stated that installation process of the leak-limiting Alloy 800 sleeve is designed to impart minimal residual stresses to the parent tube to avoid potential corrosion in the hydraulic expansion joints.

Westinghouse has performed various corrosion tests and assessments of leak-limiting Alloy 800 sleeves with full length sleeved tube mock-ups. Sleeve/tube assemblies were pressurized with highly corrosive solutions. Westinghouse also performed the corrosion tests to assess the relative time to cracking of the sleeve/tube joint. Leak-limiting Alloy 800 sleeves did not develop any cracking in either the primary or secondary side tests. The leak-limiting Alloy 800 sleeve has demonstrated higher corrosion resistance than the Alloy 600 parent tube.

Westinghouse stated that the leak-limiting Alloy 800 sleeves have not experienced service-induced degradation or leakage in nuclear plants. Westinghouse 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. This is based on experience of over 200,000 tubes in service. The NRC staff notes that, in recent years, some indications of potential SCC have been found in Alloy 800 tubing in international plants. The NRC staff also notes that the time for the initiation of corrosion in sleeve/tube assemblies is difficult to accurately quantify. Although vendors traditionally conduct accelerated corrosion tests of sleeve/tube assemblies to predict service life, the NRC staff finds this method unreliable for deterministic predictions. While the NRC staff does consider the corrosion tests to give a viable indicator of potential performance, at present, the NRC staff can only assume a limited life expectancy for leak-limiting Alloy 800 sleeves. Considering the uncertainties in sleeve life expectancy, sleeves are periodically inspected to ensure any flaws in the sleeve/tube assembly are detected and addressed. The NRC staff finds this acceptable, since it should ensure the timely detection of any degradation should it occur.

3.5 Sleeve Inspection

The licensee proposed a TS requirement to perform an inspection of the parent tube, in the area where the sleeve-to-tube lower hard roll joint and the sleeve-to-tube upper hydraulic expansion 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 transitions 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 (EDM) 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. As a result of NRC staff questions pertaining to the inspectability 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. In the licensee's October 10, 2008 letter, the licensee included a Westinghouse report titled, "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 OD EDM flaws, with depths ranging from 40 to 100 percent through-wall (TW), in the parent tube behind the nickel band of the Alloy 800 sleeves. The report also concluded that OD stress-corrosion cracking flaws that are 100 percent TW, and approaching 100 percent TW, were readily detectable at this location using current examination techniques. The licensee 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.

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 event 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.

3.6 Sleeve Structural Analysis

Westinghouse performed structural analyses 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. In the analyses, Westinghouse assumed two bounding tube configurations: (1) the tube is intact, and (2) the tube is severed at the flaw location. In addition, Westinghouse assumed two bounding tube support plate configurations: (1) the tube is free to move past the tube support plates, and (2) the tube is locked in the first tube support plate 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.

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 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 °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 staff since the overall behavior of the sleeve/tube assembly will not change.

3.7 Sleeve Leakage Integrity

Westinghouse has determined the sleeve joint leakage via 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.

3.8 SG Tube Inspection Report

The licensee stated that they would change some wording in TS Section 5.6.6.2.4 regarding the submission of an inspection report following an outage in which the F* methodology was applied. The change makes this section of the TSs consistent with the format used in Sections 5.6.6.2.1 and 5.6.6.2.2, and clarifies that the reported information involves only inspection results from the hot-leg tubesheet region where the F* methodology was applied. This change is acceptable to the NRC staff since it is an editorial change that clarifies the requirements.

3.9 NRC Staff Findings

The NRC staff concludes that the use of leak limiting Alloy 800 sleeves as a repair method for SG tubes is acceptable for the reasons stated above.

4.0 REGULATORY COMMITMENTS

Should a SG tube inspection reveal an inwardly deformed Alloy 800 sleeve, the licensee will report this to the NRC within 45 days. The special report of the Alloy 800 sleeve incident will include a root cause analysis, an assessment of the integrity of the sleeve/parent tube complex and identification of the associated corrective actions.

5.0 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.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 (74 FR 7482). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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: September 30, 2009

September 30, 2009

Mr. Peter P. Sena III
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 NO. 2 - ISSUANCE OF AMENDMENT RE: THE USE OF WESTINGHOUSE LEAK-LIMITING ALLOY 800 SLEEVES FOR STEAM GENERATOR TUBES REPAIR (TAC NO. MD9969)

Dear Mr. Sena:

The Commission has issued the enclosed Amendment No. 170 to Facility Operating License No. NPF-73 for the Beaver Valley Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 10, 2008, as supplemented by letters dated June 16 and July 14, 2009.

The amendment revises TS 5.5.5 to allow an additional method of repair for steam generator (SG) tubes by installation of leak limiting Alloy 800 sleeves developed by Westinghouse and clarifies an existing reporting requirement in TS 5.6.6.2.4 concerning SG tube inspections.

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

Sincerely,

/ra/

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-412

Enclosures:
1. Amendment No. 170 to NPF-73
2. Safety Evaluation

cc w/encls: Distribution via Listserv

ADAMS Accession No. ML092590189 *Input received. No substantive changes made.

OFFICE	LPL1-1/PM	LPL1-1/LA	CSGB/BC	ITSB/BC	OGC	LPL1-1/BC
NAME	NMorgan	SLittle	MGavrilas	RElliot (GWaig for)	LSubin (NLO w/ comments)	NSalgado (JBoska for)
DATE	9/24/09	9/21/09	8/14/2009*	9/24/09	9/25/09	9/30/09

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