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

April 6, 2012

Mr. Paul A. Harden
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 – CORRECTION LETTER TO THE REVISED STEAM GENERATOR INSPECTION SCOPE USING F* INSPECTION METHODOLOGY AMENDMENT (TAC NO. ME3498)

Dear Mr. Harden:

By letter dated February 24, 2011, the Nuclear Regulatory Commission (NRC) issued Amendment No. 172 to Renewed Facility Operating License (FOL) No. NPF-73 for the Beaver Valley Power Station Unit, No. 2 to modify the Technical Specifications to revise the scope of the steam generator (SG) tube inspections for the portion of the tube in the tubesheet on the cold-leg side of the SG by using the F* methodology.

For clarification purposes, the NRC staff would like to make some corrections to the amendment. Enclosed are the corrected pages. If you have any questions, please contact me at (301) 415-1016.

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

Enclosure:
As stated

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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 RENEWED FACILITY OPERATING LICENSE

Amendment No. 172
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. (the licensee), dated February 26, 2010, as supplemented by letters dated November 30, 2010, and January 26, 2011, 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 Renewed Facility Operating License No. NPF-73 is hereby amended to read as follows:

Corrected by letter dated, Friday, April 6, 2012

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO RENEWED

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 February 26, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML100630422), as supplemented by letters dated November 30, 2010 (ADAMS Accession No. ML103370240) and January 26, 2011 (ADAMS Accession No. ML110320242), FirstEnergy Nuclear Operating Company (FENOC, the licensee), requested changes to the Technical Specifications (TSs) for Beaver Valley Power Station, Unit No. 2 (BVPS-2). The supplements dated November 10, 2010, and January 26, 2011, 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 January 11, 2011 (76 FR 1648).

The changes would revise the scope of the steam generator (SG) tube inspections for the portion of the tube in the tubesheet on the cold-leg side of the SG by using the F* methodology. Previously, the NRC staff approved the use of the F* methodology for use on the hot-leg side of the SGs by letter dated September 27, 2006 (ADAMS Accession No. ML062580419). The existing TS 5.5.5.2.c.1 states that, "tubes found by inservice inspection (ISI) 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 inservice through application of the alternate repair criteria discussed in Specification 5.5.5.2.c.4 or 5.5.5.2.c.5." TS 5.5.5.2.c.5 currently specifies that the 40 percent depth criterion for tube repair does not need to be applied in the hot-leg tubesheet region below the "F* distance" in the tubesheet. The license amendment request (LAR) adds the cold-leg tubesheet region to hot-leg region already

Corrected by letter dated, Friday, April 6, 2012 specified in TS 5.5.5.2.c.5, and also prohibits application of the F* methodology in the tubesheet region where laser or tungsten inert gas welded sleeves have been installed. According to the F* methodology in TS 5.5.5.2.c.5, flaws below the F* distance may remain in service regardless of size. Implementing the F* methodology also eliminates the need to inspect the portion of the tube within the hot-leg and cold-leg tubesheet regions below this specified distance, since the inspection provision in TS 5.5.5.2.d requires that tubes be inspected with the objective of detecting flaws that may satisfy the applicable tube repair criteria. With no repair criteria to satisfy, the portion of the tube below the specified distance is not subject to the inspection provision. The proposed amendment adds reporting of the cold-leg tubesheet region inspection results (associated with the F* methodology) to the reporting requirement of the hot-leg tubesheet region inspection results, currently required in TS Section 5.6.6.2.4.

2.0 REGULATORY EVAULATION

2.1 Description of System

Steam generator tubes function as an integral part of the reactor coolant pressure boundary and, in addition, serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. Because of the importance of SG tube integrity, the NRC requires the performance of periodic ISIs of SG tubes. These inspections detect, in part, flaws in the tubes resulting from interaction with the SG operating environment. ISIs may also provide a means of characterizing the nature and cause of any tube flaws so that corrective measures can be taken. Tubes with flaws that exceed the tube repair criteria specified in a plant's TS are removed from service by plugging or are repaired by sleeving. The TSs provide the acceptance criteria related to the results of SG tube inspections.

The requirements for the inspection of SG tubes are intended to ensure that this portion of the reactor coolant system maintains its integrity. Tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis. Tube integrity includes both structural and leakage integrity. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubes. Leakage integrity refers to limiting primary-to- secondary leakage during normal operation, plant transients, and postulated accidents. These limits ensure the radiological dose consequences associated with any leakage are within acceptable limits and they limit the frequency of SG tube ruptures.

2.2 Regulatory Requirements and Guidance

In reviewing requests of this type, the NRC staff verifies that a methodology exists that maintains the structural and leakage integrity of the tubes consistent with the plant design and licensing basis. This includes verifying that the applicable General Design Criteria (GDC), e.g., GDCs 14 and 32, contained in Appendix A of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR) and the performance criteria in the plant TSs are satisfied. The NRC staff's evaluation is based, in part, on ensuring that the structural margins inherent in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [pressurized-water reactor] SG Tubes," are maintained. The NRC staff's evaluation also includes verifying that a conservative methodology exists for determining the amount of primary-to-secondary leakage that may occur during design-basis accidents (DBAs). The amount of leakage is limited to ensure that offsite and control room dose criteria are met. The radiological dose criteria are specified, in part, in 10 CFR Part 100, in 10 CFR 50.67, and in GDC 19 of Appendix A to 10 CFR Part 50.

The NRC approved a similar redefinition of a tube inspection for the original SGs at the Kewaunee Power Station in 1996 (NUDOCS 9609230197), for the Joseph M. Farley Plant, Unit 2 (Farley Unit 2) in 1996 (NUDOCS 9610220228), for the Comanche Peak Steam Electric Station, Unit 1 in 1999 (NUDOCS 9909030072), for the Watts Bar Nuclear Plant, Unit 1 in 2000 (ADAMS Accession No. ML003748725) and others. In each case, plant-specific repair criteria were determined.

3.0 TECHNICAL EVALUATION

3.1 Background

BVPS-2 is a 3-loop, Westinghouse-designed plant with Model 51M SGs. Each SG contains 3376 mill-annealed Alloy 600 tubes with an outside diameter of 0.875-inch and a wall thickness of 0.050-inch. The tubes in each SG are supported by horizontal support plates with drilled holes. All tube support material is carbon steel. The tubes were expanded with a mechanical rolling process (hardroll) at both ends for the full length of the tubesheet (21 inches). A weld joins the tube end to the cladding on the primary face of the tubesheet, providing a leak-tight boundary and resistance to tube pullout. The hardroll process produces an interference fit between the tube and tubesheet which can also provide resistance to tube pullout. The transition from the expanded portion of the tube to the unexpanded portion of the tube is referred to as the roll transition. Prior to operation, the internal surfaces of the tubes within the tubesheet on the hot-leg and cold-leg sides of the tubesheet were shotpeened, which applies a compressive stress that generally increases resistance to stress-corrosion cracking. The existing TSs for BVPS -2 permit the installation of three types of sleeves in order to repair flaws. The sleeves have both upper and lower joints that form the interface with the parent tube.

The tube-to-tubesheet joint consists of the tube, which is roll-expanded against the bore of the tubesheet, the tube-to-tubesheet weld located at the tube end, and the tubesheet. Typically, plants designed the tube-to-tubesheet joint as a welded joint rather than a friction or expansion joint. That is, the weld itself was designed as a pressure boundary element, and it was designed to transmit the entire end cap pressure load during normal and DBA conditions from the tube to the tubesheet with no credit taken for the friction developed between the roll-expanded tube and the tubesheet. In addition, the weld makes the joint leak tight.
The existing inspection and repair requirements in the plant TSs do not take into account the reinforcing effect of the tubesheet on the external surface of the expanded tube. Nonetheless, the presence of the tubesheet constrains the tube and complements tube integrity in that region, by preventing tube deformation beyond the expanded outside diameter of the tube. The resistance to both tube rupture and tube collapse is significantly enhanced by the tubesheet reinforcement. In addition, the proximity of the tubesheet to the expanded tube significantly reduces the leakage from any through-wall defect.

Based on these considerations, power reactor licensees have proposed, and the NRC has approved, alternate repair criteria for SG tube defects located in the lower portion of the tubesheet, when these defects are a specific distance below the expansion transition or the top of the tubesheet (TTS), whichever is lower. The F* methodology defines a distance, referred to as the F* distance, such that any type or combination of flaws below this distance (including flaws in the tube-to-tubesheet weld) is considered acceptable. That is, even if inspections below the F* distance identify flaws, the regulatory requirements pertaining to tube structural and leakage integrity would be met provided there were no significant flaws within the F* distance. The F* distance is measured from the TTS or the bottom of the roll transition (BRT), whichever is lower.

Determination of the F* distance includes a nondestructive examination (NDE) uncertainty value of 0.25 inch, which was established in the F* evaluation for Farley Unit 2 and subsequently approved as part of the staff's safety evaluation for that repair criteria. It also includes an adjustment for the location of the BRT in relation to the TTS. The value of F* calculated for structural and leakage integrity, without adjustments for NDE uncertainty and BRT location, is called the F* length. That is, the F* distance is the sum of the F* length, the NDE uncertainty, and the BRT adjustment.

The F* evaluation presented in WCAP-16385, "F* Tube Plugging Criterion For Tubes With Degradation In The Tubesheet Roll Expansion Region Of The Beaver Valley Unit 2 Steam Generators," Revision 1 (ADAMS Accession No. ML051040084), was performed for the expected operating conditions at BVPS-2 (including an 8-percent extended power uprate (EPU) which was subsequently approved by the NRC on July 19, 2006) and for DBAs. The F* value determined for the limiting faulted condition (SG feedwater line break (FLB)) bounds the current normal operating conditions and EPU conditions, with up to 22 percent tube plugging.

The F* analysis considered the forces acting to pull the tube out of the tubesheet (i.e., from the internal pressure in the tube) and the forces acting to keep the tube in place. These latter forces are a result of friction and the forces arising from (1) the residual preload from the installation (rolling) process, (2) the differential thermal expansion between the tube and the tubesheet, and (3) internal pressure in the tube within the tubesheet. In addition, the effects of tubesheet bow, due to pressure and thermal differentials across the tubesheet, were considered since this bow causes dilation of the tubesheet holes from the secondary face to approximately the midpoint of the tubesheet and reduces the ability of the tube to resist pullout. The amount of tubesheet bow varies as a function of radial position, with locations near the periphery experiencing less bow. The effects of tubesheet hole dilation were analyzed using the worst-case hole (location) in the tubesheet.

3.2 FENOC Proposal

The licensee's basis for revising the criteria for tube repair within the cold-leg tubesheet region is documented in its LAR, in WCAP-16385, Revision 1, and in the November 10, 2010, and January 26, 2011, supplemental letters. These documents also referred to WCAP-11306, "Tubesheet Region Plugging Criterion for the Alabama Power Company Farley Nuclear Station Unit 2 Steam Generators," Revision 2, April 1987, which describe the analysis and testing performed to justify a similar modification in the tube repair criteria for the Farley Nuclear Station, Unit 2.

For tubes with no portion of a lower sleeve joint in the hot-leg or cold-leg tubesheet region, TS 5.5.5.2.c.5.a specifies that the tube must be repaired or plugged if any flaw is detected within 3 inches below the TTS or 2.22 inches below the BRT, whichever elevation is lower. For tubes which have any portion of a sleeve joint in the hot-leg tubesheet region, TS 5.5.5.2.c.5.b specifies that the tube must be plugged if any flaw is detected within 3 inches below the lower end of the lower sleeve joint. Any flaw located below the elevations specified in proposed TSs 5.5.5.2.c.5.a and 5.5.5.2.c.5.b would be allowed to remain in service regardless of size.

The following sections summarize the NRC staff's evaluation of the proposed BVPS-2 F* proposal in terms of maintaining SG structural and leakage integrity.

3.3 Tube Structural Integrity

The amendment would permit tubes with flaws to remain in service; therefore, the licensee must demonstrate that the tubes kept in service using the F* methodology will maintain adequate structural integrity for the period of time between inspections. Tube rupture and pullout of a tube from the tubesheet are the two potential credible modes of structural failure considered for tubes returned to service under the F* methodology.

In order for a tube to rupture, a flaw would need to grow above the tubesheet's secondary face. If the entire flaw remains within the tubesheet, the reinforcement provided by the tubesheet will prevent tube rupture. The F* methodology proposed by the licensee for BVPS-2 requires an inspection of the top portion of the tube within the hot-leg or cold-leg tubesheet and the plugging of any flaws in this region. Therefore, any known flaws remaining in service following the inspections will be located a minimum of 3 inches below the TTS or the lower joint of a sleeve. Industry operating experience shows flaw growth rates within the tubesheet are well below those necessary to propagate a flaw from 3 inches below the TTS to outside the tubesheet in one operating cycle (typically 18 months). Therefore, it is unlikely that any of these flaws will grow in an axial direction and extend outside the tubesheet during one operating cycle. Similarly, it is unlikely that a flaw would propagate upward to a sleeve joint from 3 inches below the joint during one operating cycle. Thus, tube burst is precluded for these flaws due to the reinforcement provided by the surrounding tubesheet.

In the event that undetected flaws are present in the F* distance, or that new flaws initiate in the F* distance during the operating cycle following an inspection, it is possible that these flaws could grow in the axial direction and extend outside the tubesheet. As a result, the NRC staff considered the conditions that would be necessary to structurally fail a tube with this type of flaw. Steam generator tube rupture is primarily a function of flaw geometry, the differential pressure across the tube wall, and the flaw location. Axial through-wall flaws may result in a tube failing to maintain adequate margins for burst under all operating conditions. However, this would require the flaws to exceed a certain length, typically on the order of one-half inch or longer, and have no external restraint (i.e., occur in the free span). Partially through-wall flaws would require additional length (beyond the one-half inch postulated above) in order to become susceptible to spontaneous rupture based on empirical models for tube burst. Thus, these flaws would have to extend a significant distance above the tubesheet to degrade the margins of structural integrity for the affected tube (i.e., tubes with undetected flaws slightly below the TTS).

In addition, constraining a flaw at one end by the tubesheet would further elevate the burst pressure of this tube (compared to an identical flaw with no constraint). Flaw growth rates

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