September 27, 2006

Mr. James H. Lash
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: REVISED STEAM GENERATOR INSPECTION AND REPAIR SCOPE USING THE F* METHODOLOGY (TAC NO. MC6768)

Dear Mr. Lash:

The Commission has issued the enclosed Amendment No. 160 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 in response to your application dated April 11, 2005, as supplemented December 2, 2005, and January 27, April 14, August 16, and September 1, 2006.

The amendment revises the scope of the steam generator tubesheet inspections and subsequent repair using the F* inspection methodology.

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/

Timothy G. Colburn, Senior 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. 160 to NPF-73
2. Safety Evaluation

cc w/encls: See next page

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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 FACILITY OPERATING LICENSE

Amendment No. 160
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 April 11, 2005, as supplemented December 2, 2005, and January 27, April 14, August 16, and September 1, 2006, 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) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 160, 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 the date of its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, 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 27, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 160

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	Insert
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	Insert
6-22a	6-22a
6-23	6-23
6-28	6-28
6-29	6-29
6-31	6-31
6-32	6-32
- -	6-33

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 160 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 (BVPS-2)

DOCKET NO. 50-412

1.0 INTRODUCTION

By letter dated April 11, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML051040075), and supplemented by letters dated December 2, 2005 (ADAMS Accession No. ML053420343), January 27, 2006 (ADAMS Accession No. ML060330258), April 14, 2006 (ADAMS Accession No. ML061100182), August 16, 2006 (ADAMS Accession No. ML062300027), and September 1, 2006 (ADAMS Accession No. ML062490200), FirstEnergy Nuclear Operating Company (FENOC), the licensee for BVPS-2, requested changes to the Technical Specifications (TSs). The supplements dated December 2, 2005, and January 27, April 14, August 16, and September 1, 2006, 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 or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 7, 2005 (70 FR 33214). The Commission's issuance of Amendment No. 158 to Facility Operating License NPF-73 for BVPS-2, regarding steam generator tube integrity (Technical Specification Task Force (TSTF) Item 449) on September 7, 2006, resulted in renumbering and rewording the requirements as originally proposed by the licensee to fit the TSTF-449 format, but did not change the scope of the application.

The proposed amendment revises the scope of the steam generator (SG) tubesheet inspections and repair using the F* (F-star) inspection methodology. Specifically, the proposed amendment would revise the BVPS-2 TSs to change the requirements for SG tube inspection and repair in the SG hot-leg tubesheet region by applying a methodology called F*. The F* methodology was developed for plants with tubes that were expanded into the tubesheet region with a mechanical roll process. The existing BVPS-2 TS 6.19.c.1 specifies that tubes containing flaws equal to or greater than 40 percent in depth must be plugged or repaired by sleeving unless an exception applies. The proposed amendment adds TS 6.19.c.5, which specifies that the 40-percent depth criteria for tube repair does not need to be applied in the hot-leg tubesheet region below a certain elevation in the tubesheet. According to the F* methodology in TS 6.19.c.5, flaws below this elevation 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 tubesheet region below this elevation, since the inspection provision in TS 6.19.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 this elevation is not subject to the inspection provision.

The proposed change will also add TS 6.19.d.5 to require inspection of 100 percent of the tubes over a certain distance in the upper part of the hot-leg tubesheet region, and TS 6.9.7.4 to include new reporting requirements associated with implementing the F* methodology.

2.0 REGULATORY EVALUATION

SG 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 inservice inspections of SG tubes. These inspections detect, in part, flaws in the tubes resulting from interaction with the SG operating environment. Inservice inspections 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 TSs are removed from service by plugging or are repaired by sleeving. The BVPS-2's 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.

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 1.121, "Bases for Plugging Degraded PWR [pressurized-water reactor] SG Tubes," are maintained. The 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 approximately 3390 mill-annealed Alloy 600 tubes with an outside diameter of 0.875 inches and a wall thickness of 0.050 inches. 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 on the hot-leg side 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 two types of sleeves in order to repair flaws. Both sleeve types have an upper and lower joint that forms 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 essentially precluding 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 non-destructive examination (NDE) uncertainty value of 0.25 inches, 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. ML051040081), 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 hot-leg tubesheet region is documented in its license amendment request, in WCAP-16385, Revision 1, and in its supplemental letters listed above. These documents also referred to WCAP-14697, "L* Tube Plugging Criteria For Tubes With Degradation In The Tubesheet Roll Expansion Region Of The Farley 2 Steam Generators," July 1996, and 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 tubesheet region, proposed TS 6.19.c.5.a specifies that the tube must be repaired or plugged if any flaw is detected within 3 inches below the TTS or 2.2 inches below the BRT, whichever elevation is lower. For tubes which have any portion of a sleeve joint in the hot-leg tubesheet region, proposed TS 6.19.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 6.19.c.5a and 6.19.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 proposed 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 the 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 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 below 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 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. SG tube rupture is primarily a function of flaw geometry (e.g., length), 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 necessary for undetected or newly initiated flaws to reach a critical flaw size are unlikely to occur given the inspections that are required to be performed. Therefore, flaws remaining in service under either of the two scenarios described above should result in the tube(s) maintaining adequate margins for tube burst.

The other postulated structural failure mode for tubes remaining in service using the F* methodology is pullout of the tube from the tubesheet due to axial loading on the tube. Differential pressures from the primary side to the secondary side of the SG impart axial loads into each tube that are reacted at the tube-to-tubesheet interface. Axial tube loading during normal operating conditions can be significant. The peak postulated loading, however, occurs during events involving a depressurization of the secondary side of the SG, such as an FLB or main steamline break. The presence of flaws within an SG tube decreases the load-bearing capability of the affected tube. If a tube becomes sufficiently degraded, these loads could lead to an axial separation of the tube.

The analysis supporting the licensee's proposed modifications to the tube inspection requirements addressed the limiting conditions necessary to maintain adequate structural integrity of the tube-to-tubesheet joint. Specifically, the tube must not experience excessive displacement relative to the tubesheet under bounding loading conditions with appropriate factors of safety considered. Safety factor criteria are derived from the ASME Code, Section III, and are a comparison of applied stresses to the ultimate strength of the tube material. For F*, the most limiting condition for structural integrity is maintaining a margin (safety factor) of 1.4 against the axial loads experienced during faulted conditions.

To justify the structural integrity acceptability of any flaw or combination of flaws below the F* distance, the licensee completed an assessment using analytical calculations and laboratory experiments. This assessment included measurements of the elastic radial preload due to the hardrolling process using tube sections rolled into simulated tubesheets (collars). Physical dimensions were measured before and after rolling the tube sections into the collars, and then again after removing the collar. The amount of tube deflection was analyzed to determine the amount of preload radial stress present following the rolling process. The assessment also included calculations of the changes in radial preload during operation due to thermal expansion tightening, differential pressure, and tubesheet bow for normal operating and faulted conditions. The required engagement distance, F*, was then calculated by equating the load-carrying ability of the tube (preload frictional forces) to the applied operating loads. These calculations included a reduction in the load-carrying ability near the ends of a severed tube.

The F* values calculated for current normal operating, normal operating after power uprate, and faulted SG conditions were 1.74 inches, 1.77 inches, and 1.97 inches, respectively. These values were determined using a safety factor of 3 for normal operating conditions and 1.4 for faulted conditions. The most limiting of these values was used to specify the F* length of 1.97 inches in determining the required engagement length of tubing.

In summary, the use of the F* methodology will (1) limit the potential for the growth of flaws in the tubesheet region into the freespan region above the tubesheet, and (2) ensure the tubes will not pull out of the tubesheet. On these bases, the NRC staff has concluded that tubes returned to service using the F* repair criteria will maintain adequate structural integrity.

3.4 Tube Leakage Integrity

In assessing leakage integrity of an SG under postulated accident conditions, the leakage from all sources (i.e., all types of flaws at all locations and all non-leak tight repairs) must be assessed. The combined leakage from all sources is limited to below a plant-specific limit based primarily on radiological dose consequences. This limit is referred to as the "accident-induced leakage limit." The licensee's approach to addressing leakage from flaws within the tubesheet region considers two regions: (1) the upper portion of the tube within the hot-leg tubesheet, within 3 inches below the TTS or within 2.2 inches below the BRT, whichever is greater, and (2) the region more than 3 inches below the TTS or 2.2 inches below the TTS, whichever is greater. In general, the licensee assumes there will be no leakage from either region. As discussed below, the NRC staff determined that the leakage from either region will not be significant.

In the top part of the tubesheet, the region in which all tubes with detected flaws must be plugged or repaired, operating experience suggests it is unlikely that through-wall (or near through-wall) flaws will develop given that this area is inspected at least once every 24 effective full-power months. However, the licensee stated that if flaws are detected they will be evaluated for their effect on the leakage integrity of the SG to confirm this expectation.

For flaws below the region in which all tubes with detected flaws must be plugged or repaired, the licensee's evaluation considered the effects of the hardroll installation, the primary-to-secondary pressure differential, differential thermal expansion, and tubesheet bow on the interference fit between the tube and tubesheet, and compared these effects with the leakage driving force from the primary-to-secondary pressure differential. As discussed above, the evaluation included measurements of the elastic radial preload from the hardrolling process using tube sections rolled into simulated tubesheets (collars). These tests indicated that at 3 inches and greater below the TTS (or 2.2 inches and greater below the BRT), the contact pressure for all tubes in the hot-leg tubesheet region will be higher than the highest anticipated internal pressure of 2650 psi corresponding to an FLB.

Tests to estimate the amount of leakage from a tube with a 360-degree, through-wall circumferential crack within the tubesheet region were performed as part of the evaluation for similar inspection and repair criteria developed for other plants and documented in WCAP-14697. These tests consisted of tubes rolled into steel collars simulating the tubesheet, pressurized to various levels using water at elevated temperature. The tubes had through-wall holes around the entire circumference to simulate the flaw. These simulated flaws were conservative representations of actual cracks. Because of their geometry, actual cracks can be expected to restrict flow more than the simulated (i.e., drilled hole) flaws.

At 619 degrees Fahrenheit and a test pressure of 2650 psi (associated with faulted conditions), leakage was detected in three of five specimens with roll expansion lengths of 1or 2 inches. The maximum leak rate of these three specimens was 1.1×10^{-4} gpm. The licensee and the NRC staff calculated different average leak rates for these tests (3.1×10^{-5} gpm and 2.5×10^{-5} gpm, respectively). It appears to the staff that the higher calculated average leak rate excludes one of the samples with zero leakage. Nonetheless, this low rate of leakage, coupled with the low likelihood of developing a significant number of through-wall flaws near the TTS (approximately 3 inches below), indicates flaws in this region will not be a significant leakage source relative to the plant's leakage limit. In addition, the shotpeening of the roll-expanded tubes should further reduce the likelihood of developing a significant number of severe flaws.

In summary, the NRC staff concludes that the proposed tube-to-tubesheet joint length (or inspection distance) is acceptable to ensure that the amount of accident-induced leakage from undetected flaws below the F* distance (i.e., below the inspection distance) will be negligible compared to the leakage rate assumed in the licensee's accident analyses.

The NRC has previously approved similar F* amendments for other plants that assumed negligible accident-induced leakage.

3.5 Reporting Requirements

As part of implementation of the F* methodology, the licensee would create TS 6.9.7.4 to require specific information to be submitted to the NRC within 90 days after the reactor coolant system achieves Mode 4 following the outage in which the F* methodology was applied. The following information will be reported under this TS requirement:

- Total number of flaw indications
- Location, orientation, and severity of each indication
- Tube surface (inside or outside) from which each indication initiated
- Cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet
- The projected end-of-cycle accident-induced leakage from tubesheet indications

This report will permit the NRC staff to verify the operating experience continues to be conservative relative to the assumptions made in the amendment. As a result, the staff concludes that the proposed changes to the TS reporting requirements are acceptable.

3.6 Summary

The NRC staff finds the licensee's proposed methodology for assessing structural and leakage integrity for flaws in the tubesheet region acceptable. Therefore, the staff concludes that the licensee's proposed repair criteria are acceptable (including inspection and reporting requirements).

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 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 (70 FR 33214). 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.

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) 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.

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