FENOC

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April 11, 2005
L-05-061

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
License Amendment Request No. 183
Revised Steam Generator Inspection Scope

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the operating license for Beaver Valley Power Station (BVPS) Unit No. 2 in the form of changes to the Technical Specifications. The proposed amendment will revise the scope of the steam generator tubesheet inspections using the F* methodology. This letter satisfies a commitment made by FENOC in Letter L-04-036, dated October 28, 2004, in which it was stated that a license amendment request would be submitted to modify the definition of "tube inspection" to be consistent with Generic Letter 2004-01.

The FENOC evaluation of the proposed changes are presented in an enclosure. The enclosure contains proposed Technical Specification changes in Attachment A. Attachment B provides proposed "information-only" changes to the Technical Specification Bases that reflect the proposed license amendment. Attachment C provides one copy of WCAP-16385-NP, Revision 1, "F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of Beaver Valley Unit 2 Steam Generators," dated March 2005 (non-proprietary). Attachment D provides one copy of WCAP-16385-P, Revision 1, " F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of Beaver Valley Unit 2 Steam Generators," dated March 2005 (proprietary).

The Beaver Valley review committees have reviewed the changes. The changes were determined to be safe and do not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the enclosed evaluation of the proposed change.

FENOC requests approval of the proposed amendment by July 1, 2006, in order to support implementation of the proposed changes for the BVPS Unit 2 refueling outage in the fall of 2006. Once approved, the amendment shall be implemented within 60 days.

A001

Enclosed is Westinghouse authorization letter, CAW-05-1972 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

As Attachment D to the enclosed evaluation of the proposed change contain information proprietary to Westinghouse Electric Company LLC, it is supported by the affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-05-1972 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

There are no new regulatory commitments contained in this letter. If there are any questions concerning this license amendment request, please contact Mr. Henry L. Hegrat, Supervisor, Licensing at 330-315-6944.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 11, 2005.

Sincerely,

L. W. Pearce

L. William Pearce

Enclosures:

- FENOC Evaluation of the Proposed Change
- Westinghouse authorization letter, CAW-05-1972, dated March 24, 2005
- Proprietary Information Notice/Copyright Notice

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Sr. Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP (w/o Attachment D)
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Central File - *Keywords: LAR, Steam Generator, F*, Tubesheet*

ENCLOSURE

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 2A-183

FENOC Evaluation of the Proposed Change

Subject: Application for Amendment of Technical Specification 3/4.4.5 to revise the Steam Generator Inspection Scope.

Table of Contents

Attachments

Number	Title
A	Proposed Technical Specification Changes
B	Proposed Technical Specification Bases Changes
C	WCAP-16385-NP, Revision 1, "F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Beaver Valley Unit 2 Steam Generators," dated March 2005 (Non-proprietary)
D	WCAP-16385-P, Revision 1, "F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Beaver Valley Unit 2 Steam Generators," dated March 2005 (Proprietary)

1.0 DESCRIPTION

FirstEnergy Nuclear Operating Company (FENOC) requests amendment of Operating License NPF-73 for Beaver Valley Power Station Unit No. 2 (BVPS-2). The proposed amendment would revise Technical Specification 3.4.5, "Steam Generators," to change the scope of the steam generator (SG) tube sheet examinations required in the SG tube sheet region using the F* (F Star) inspection methodology.

FENOC is proposing to modify the BVPS-2 Technical Specifications (TSs) to revise surveillance requirements (SRs) 4.4.5.4.a.6 and 4.4.5.4.a.8 and to add SRs 4.4.5.2.e and 4.4.5.4.a.11. Technical specification SRs define SG tube inspection scope. FENOC's proposed change alters the tube inspection to exclude the portion of the tube within the tubesheet below the F* distance and to exclude the tube-to-tubesheet weld, by crediting the methodology described by WCAP-16385, Revision 1 (References 1 and 2). The F* distance is the distance from the top of the tubesheet to the bottom of the F* length (the maximum length of tubing below the bottom of the roll transition {BRT} which must be demonstrated to be non-degraded and is defined as 1.97 inches on the hot leg side) plus the distance to the BRT and non-destructive examination (NDE) uncertainties. FENOC's proposed change also revises SRs to require tubes with service-induced degradation identified in the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, to be repaired or removed from service upon detection.

2.0 PROPOSED CHANGE

The proposed changes to the BVPS-2 Technical Specification will revise the definition of SG tube inspection with respect to the tubesheet region. In this region, tube inspection would include only the portion of the tube within the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater. Tube inspection would not include the tube-to-tubesheet weld. The revised inspection scope would continue to ensure the safe operation of the plant while limiting unnecessary radiation exposure to plant personnel.

Specifically, the proposed change will revise the SR 4.4.5.4.a.8 definition for "Tube Inspection" to include only the portion of the tube in the tubesheet region that is within the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater. The proposed change will also revise SR 4.4.5.4.a.6 on SG tube repair criteria to require tubes with any service-induced degradation within the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, to be repaired or removed from service. Service induced degradation below this region in the tubesheet would be acceptable for continued operation. SR 4.4.5.2.e would be added to require rotating pancake coil inspection of the hot leg tubesheet F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, for 100 percent of the tubes sampled. New F* terminology definitions would be added in 4.4.5.4.a.11.

The proposed technical specification change, which is submitted for Nuclear Regulatory Commission (NRC) review and approval, is provided in Attachment A. The changes proposed to the Technical Specification Bases are provided in Attachment B. The proposed Technical Specification Bases changes do not require NRC approval. The BVPS Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. The Technical Specification Bases changes are provided for information only. The F* methodology is described in References 1 and 2, respectively (Attachments C (non-proprietary) and D (proprietary), respectively). These reports detail the analyses and testing performed to verify the adequacy of the F* methodology.

The proposed change to the Technical Specifications and Technical Specification Bases have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added.

To meet format requirements the Index, Technical Specifications and Bases pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

3.0 BACKGROUND

The SGs at BVPS-2 are Westinghouse Model 51M with a U-tube configuration. Each tube is secured in the tubesheet above the lower plenum of the SG by a mechanical roll process. The mechanical roll process expands each tube over its entire length within the tubesheet and forms an interference fit between the tube and tubesheet. This interference fit structurally supports the tube and provides a leak-tight boundary between the primary and secondary systems at each end of the SG tube. The tube is also welded to the tubesheet at each end. The region at the top of the tubesheet where a tube transitions between its expanded diameter and its nominal diameter is referred to as the roll expansion transition region.

A tube plugging criterion (referred to as F*) has been developed by Westinghouse Electric Company LLC (Westinghouse) to permit tubes with observed and/or postulated degradation in the mechanical roll tubesheet expansions below the F* distance to remain in service. The F* analysis defines an undegraded F* length that assures adequate strength is available to resist the axial pullout loads experienced within the tubesheet and ensure leakage integrity. This proposed change would define the Technical

Specification SG tube inspection scope to exclude the length of tubing below the F* length on the hot leg side but is not requesting an alternate repair criterion.

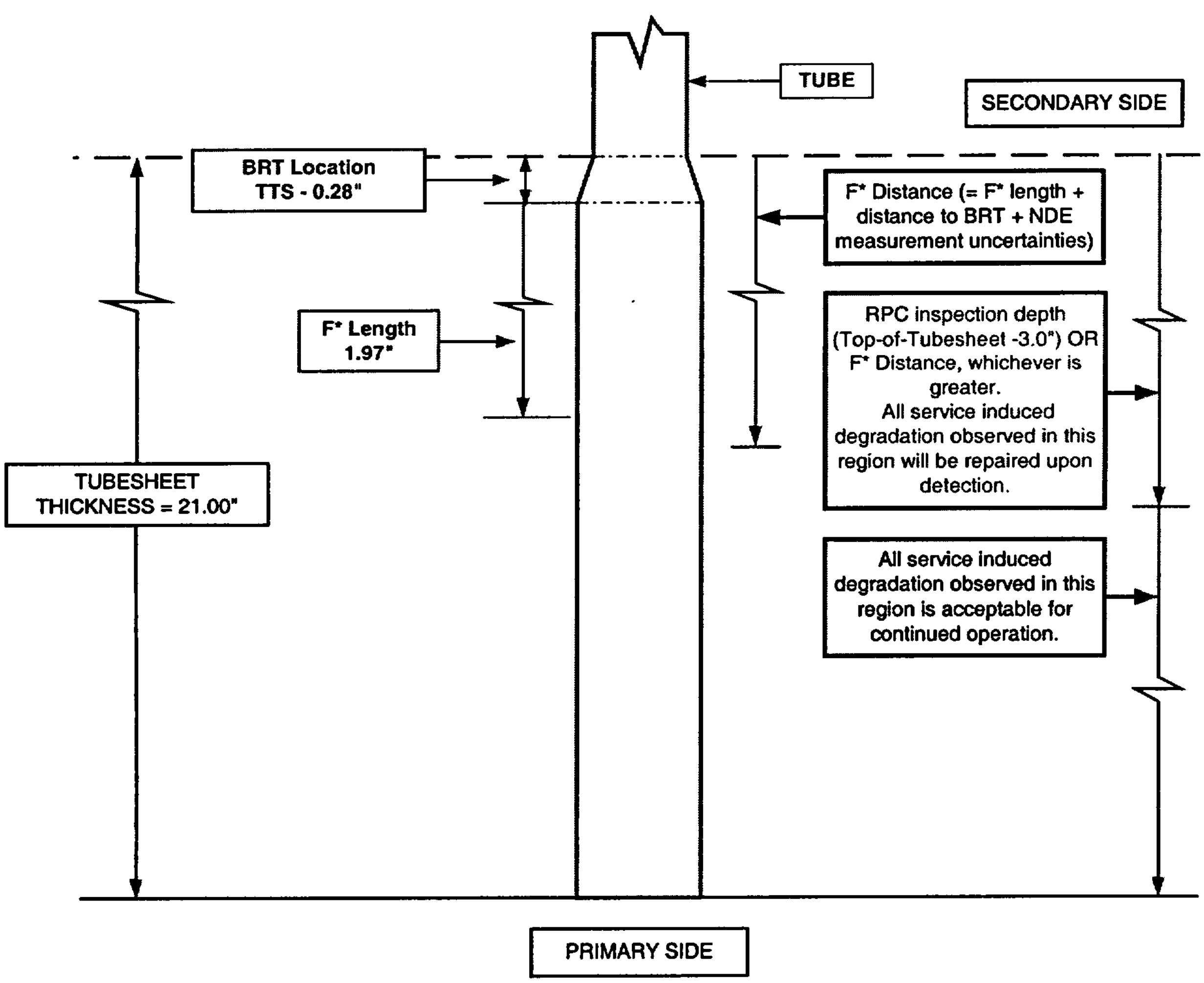
Definitions:

BRT - Bottom of the roll transition and is defined in References 1 and 2, Section 3.0, as a nominal 0.28 inches from the top of the tubesheet.

F* Length - The maximum length of tubing below the BRT which must be demonstrated to be non-degraded and is defined in References 1 and 2, Section 2.2.3, as 1.97 inches below the BRT.

F* distance - The distance from the top of the tubesheet to the bottom of the F* length including the distance to the BRT and NDE uncertainties. Uncertainties are defined in References 1 and 2, Section 2.5, as 0.25 inch.

Sketch of F* Distance in BVPS-2 Tubesheet



4.0 TECHNICAL ANALYSIS

FENOC's proposed change revises the tube inspection definition with respect to the tubesheet region. In this region, tube inspection would include only the portion of the tube within the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater. Tube inspection would not include the tube-to-tubesheet weld. It adds a requirement that tubes with service-induced degradation identified in the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, shall be repaired or removed from service upon detection. Service induced degradation below this region in the tubesheet would be acceptable for continued operation. The proposed change is based on methodology described in References 1 and 2 that was developed by Westinghouse Electric Company LLC. References 1 and 2 were developed to demonstrate the applicability of the methodology documented in WCAP-11306, Revision 2 (Reference 3) to the BVPS-2 steam generators. Reference 3 documented the F* alternate repair criteria methodology and analysis that was approved by the NRC for the Farley Unit 2 steam generators, that accounted for the reinforcing effect that the tubesheet has on the external surface of the SG tube within the tubesheet region. This analysis showed that tube integrity and leakage below the F* distance remained within the existing design limits. Reference 3 was approved by the Nuclear Regulatory Commission for Farley Unit 2 in License Amendment 64. [Operating License No. NPF-8, Docket No. 50-364, ML013130645]

Using BVPS-2 operating conditions, References 1 and 2 define the maximum F* length for pullout resistance as 1.97 inches below the bottom of the roll transition. This distance is increased by an allowance for NDE uncertainties in measuring the F* length. The maximum NDE uncertainty on the F* length in References 1 and 2 is 0.25 inches. The required inspection distance below the top of the tubesheet is then 2.50 inches which includes a 0.28 inch distance from the top-of-tubesheet to the bottom of the roll transition. The 0.28 inch distance represents the lower 95% confidence BRT value for the BVPS-2 steam generators as determined using eddy current data for all active tubes obtained during the ninth refueling outage. The F* analysis detailed in Reference 3 provides the basis for tubes with any form of degradation below the F* length to remain in service. The presence of the surrounding tubesheet prevents tube rupture and provides resistance against axial pullout loads during normal and accident conditions including the planned uprated power level of 2910 MWt. In addition, any primary-to-secondary leakage from tube degradation below the F* length is so minimal for postulated SLB event conditions that it will not affect offsite dose calculations and, therefore, can be neglected. A complete circumferential separation below the inspection distance would not be expected to leak at SLB conditions. Consequently, any tube degradation that may go undetected in the area below the inspection distance would not affect structural or leakage integrity. An inspection distance of 3.0 inches (rather than

2.5) below the top of the tubesheet has been included in the proposed technical specifications for conservatism.

Currently, BVPS-2 SG inspection fulfills TS 4.4.5.4.a.8 requirements for inspecting SG tubing by performing 100 percent full-length inspection of each tube using a bobbin coil probe. To reduce the probability and consequences of SG tube rupture or tube failure, BVPS-2 performs RPC probe examinations in critical regions for crack-like indications that would not be easily identified with the bobbin coil probe. These critical regions are based on a degradation assessment that defines where potential and active degradation is expected in SG tubes that could challenge structural and/or leakage integrity if the tubes were not repaired or removed from service.

The critical region of the tubes in the tube-to-tubesheet expansion in Westinghouse Model 51M SGs with mechanical roll expansions is defined as the F* length. The F* length is defined for BVPS-2 in References 2 and 3, considering the most stringent loads associated with plant operation, including transients, and accident conditions and the planned uprated power level of 2910 MWt. Below the F* distance, any degradation is acceptable.

FENOC does not propose to use References 1 and 2 as an alternate repair criterion to leave tubes degraded within the F* distance in service. Instead, the proposed amendment would use the references as the basis for defining the length of tubing that would be inspected using an RPC probe. Tubes with service-induced degradation identified in the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, would be repaired or removed from service upon detection. Service induced degradation below this region in the tubesheet would be acceptable for continued operation.

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Thus, structural criterion is satisfied by the tubesheet constraint. However, a 360-degree circumferential crack could permit severing of the tube and tube pullout from the tubesheet under the axial forces on the tube from primary to secondary pressure differentials. Section 4 of References 1 and 2 describe the testing that was performed to define the length of non-degraded tubing that is sufficient to compensate for the axial forces on the tube and thus prevent pullout. References 1 and 2 detail the effects of the differences in operating conditions utilized in Reference 3 for Farley Unit 2 and the operating conditions defined for BVPS-2 on the F* engagement length.

Operating experience has demonstrated negligible normal operating leakage from PWSCC even under near free span conditions in roll transitions. PWSCC in roll expansions in the tubesheet region would be leakage limited by the tight tube-to-tubesheet crevice, increased material property strength values due to cold working, and the limited crack opening permitted by the tubesheet constraint. Steamline break conditions provide the most severe radiological consequences for postulated accidents involving loss of pressure or fluid in the secondary system. Reference 3 describes the methodology for calculating leakage for cracks left in service and the justification for neglecting the total contribution of leakage through cracks below the F* distance to steamline break consequences. Therefore, RPC inspection in the area below the F* distance is not necessary to preclude normal operating or accident induced leakage.

5.0 REGULATORY SAFETY ANALYSIS

FENOC is proposing to modify the Beaver Valley Power Station Unit 2 (BVPS-2) Technical Specification (TS) steam generator (SG) tube inspection definition with respect to the tubesheet region. In this region, tube inspection would include only the portion of the tube needed to ensure structural and leakage integrity. Tubes with service-induced degradation identified in the area requiring inspection would be repaired or removed from service upon detection. Specifically, the proposed change will revise the Surveillance Requirement (SR) 4.4.5.4.a.8 definition for "Tube Inspection" to include only the portion of the tube in the tubesheet region that is within the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater. The proposed change will also revise SR 4.4.5.4.a.6 on SG tube repair criteria to require tubes with service-induced degradation within the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, to be repaired or removed from service. Service induced degradation below this region in the tubesheet would be acceptable for continued operation. SR 4.4.5.2.e would be added to require rotating pancake coil inspection of the hot leg tubesheet F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, for 100 percent of the tubes sampled. New F* terminology definitions would be added in 4.4.5.4.a.11.

The F* distance is defined in WCAP-16385, Revision 1. The WCAP demonstrates the applicability of the F* methodology detailed in WCAP-11306, Revision 2, previously approved by the NRC for Farley. This methodology accounts for the reinforcing effect of the tubesheet on the external surface of the SG tube within the tubesheet region. Additionally WCAP-11306, Revision 2, and WCAP-16385, Revision 1, show that the tube integrity and leakage below the F* distance remain within the existing design limits. The proposed change requires that tubes with service-induced degradation identified in the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, shall be repaired or removed from service upon detection. Service induced degradation below this region in the tubesheet would be acceptable for continued operation. The revised inspection scope would continue to ensure the safe operation of the plant while limiting unnecessary radiation exposure to plant personnel. The proposed change is for the purpose of defining the inspection scope and is not requesting an alternate repair criterion.

5.1 No Significant Hazards Consideration

FENOC has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. The proposed change modifies the BVPS Unit 2 TSs to incorporate steam generator tube inspection scope based on WCAP-16385, Revision 1. Of the various accidents previously evaluated in the BVPS Unit 2 Updated Final Safety Analysis Report (UFSAR), the proposed changes only affect the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB) accident evaluation. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model 51M SGs has shown that axial loading of the tubes is negligible during an SSE.

   For the SGTR event, the required structural margins of the steam generator tubes will be maintained by the presence of the tubesheet. Tube rupture is precluded for cracks in the tube expansion region due to the constraint provided by the tubesheet. Therefore, Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," margins against burst are maintained for both normal and postulated accident conditions.

   The F* length supplies the necessary resistive force to preclude pullout loads under both normal operating and accident conditions. The contact pressure results from the tube expansion process used during manufacturing and from the differential pressure between the primary and secondary side. The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of an SGTR or SLB accident.

   The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the expansion by precluding tube deformation beyond its initial expanded outside diameter. The resistance to both tube rupture and collapse is strengthened by the tubesheet in that region. At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the F* length is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

SLB leakage is limited by leakage flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of crack face opening compared to free span indications. The total leakage (i. e., the combined leakage for all such tubes) meets the industry performance criterion, plus the combined leakage developed by any other alternate repair criteria, and will be maintained below the maximum allowable SLB leak rate limit, such that off-site doses are maintained less than 10 CFR 100 guideline values and the limits evaluated in the BVPS Unit 2 UFSAR.

Therefore, based on the above evaluation, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed changes do not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity will continue to be maintained for all plant conditions upon implementation of the F* methodology.

   The proposed changes do not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   No. The proposed changes maintain the required structural margins of the steam generator tubes for both normal and accident conditions, including the planned uprated power level of 2910 MWt. NRC Regulatory Guide (RG) 1.121 is used as the basis in the development of the F* methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC staff for meeting General Design Criteria 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For primarily axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. WCAP-16385, Revision 1, defines a length, F*, of degradation-free expanded tubing that provides the necessary resistance to tube pullout due to the pressure-induced forces (with applicable safety factors applied). Application of the F* criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the F* criteria.

Plugging of the steam generator tubes reduces the reactor coolant flow margin for core cooling. Implementation of F* methodology at Beaver Valley Unit 2 will result in maintaining the margin of flow that may have otherwise been reduced by tube plugging.

Based on the above, it is concluded that the proposed changes do not result in a significant reduction of margin with respect to plant safety as defined in the Final Safety Analysis Report Update or bases of the plant Technical Specifications.

Based on the above, FENOC concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

A review of 10 CFR 50, Appendix A, "General Design Criteria (GDC) for Nuclear Power Plants" (Reference 4), was conducted to assess the potential impact associated with the proposed changes. The following table lists the criterion potentially impacted, and an assessment of the need for a modification to the UFSAR description of BVPS-2 design conformance to the criterion.

General Design Criteria		Impact
14	Reactor coolant pressure boundary	None
15	Reactor coolant system design	None
16	Reactor containment design	None
31	Fracture prevention of reactor coolant pressure boundary	None
32	Inspection of reactor coolant pressure boundary	None

The reactor coolant pressure boundary, containment boundary and tube-bundle integrity will not be adversely affected by the implementation of the F* tube inspection scope. SG tube burst or collapse cannot occur within the confines of the tubesheet; therefore, the tube burst and collapse criteria of Regulatory Guide (RG) 1.121 are inherently met. Any degradation below the F* distance is shown by analyses and test results to be acceptable, thereby precluding an event with consequences similar to a postulated tube rupture event. Steam generator tube surveillance requirements continue to ensure that degraded tubes will be repaired or removed from service upon detection. Therefore, conformance with all applicable GDCs remain valid.

In conclusion, based on the considerations discussed above, (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.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Westinghouse Electric Company WCAP-16385-NP, Revision 1, "F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Beaver Valley Unit 2 Steam Generators," March 2005. (Non-proprietary)

2. Westinghouse Electric Company WCAP-16385-P, Revision 1, "F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Beaver Valley Unit 2 Steam Generators," March 2005. (Proprietary)

3. Westinghouse Electric Company WCAP-11306, Revision 2, "Tubesheet Region Plugging Criterion for the Alabama Power Company, Farley Nuclear Station, Unit 2 Steam Generators," April 1987.

4. Beaver Valley Power Station Unit No. 2 Updated Final Safety Analysis Report.

5. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976.

6. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

ATTACHMENT A

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 2A-183

Proposed Technical Specification Changes

The following are the only affected pages:

3/4 4-11*

3/4 4-12

3/4 4-13*

3/4 4-14*

3/4 4-14a

3/4 4-14b*

3/4 4-14c*

3/4 4-14d*

3/4 4-14e

3/4 4-14f*

3/4 4-15*

3/4 4-16*

* No changes are proposed for this page. It is provided for readability only.

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy Current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3 percent of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50 percent of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and

2. Tubes in those areas where experience has indicated potential problems, and

3. At least 3 percent of the total number of sleeved tubes in all three steam generators. A sample size less than 3 percent is acceptable provided all the sleeved tubes in the steam generator(s) examined during the refueling outage are inspected. These inspections will include both the tube and the sleeve, and

4. A tube inspection pursuant to Specification 4.4.5.4.a.8. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

5. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (4.4.5.4.a.10) shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

d. 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.

e. Implementation of the steam generator F* inspection methodology requires rotating pancake coil inspection of the hot leg tubesheet F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, for 100 percent of the tubes sampled.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5 percent of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1 percent of the total tubes inspected are defective, or between 5 percent and 10 percent of the total tubes inspected are degraded tubes.
C-3	More than 10 percent of the total tubes inspected are degraded tubes or more than 1 percent of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes or sleeves must exhibit significant (greater than 10 percent) further wall penetrations to be included in the above percentage calculations.

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under All Volatile Treatment (AVT) conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,

2. A seismic occurrence greater than the Operating Basis Earthquake,

3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

4. A main steamline or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube or sleeve from that required by fabrication drawings or specifications. Eddy-current testing indications below 20 percent of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube or sleeve.

3. Degraded Tube means a tube or sleeve containing imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation.

4. Percent Degradation means the percentage of the tube or sleeve wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:

   a) Original tube wall 40%

      1.0 This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.10 for the repair limit applicable to these intersections.

      2.0 This definition does not apply to service-induced degradation identified in the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater. Tubes with service-induced degradation identified in this region, shall be repaired or removed from service upon detection.

      3.0 This definition does not apply to service-induced degradation identified below the F* distance and also greater than 3.0 inches below the top of the tubesheet. Tubes with service-induced degradation identified in this region are acceptable for continued operation.

   b) ABB Combustion Engineering TIG welded sleeve wall 32%

   c) Westinghouse laser welded sleeve wall 25%

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steamline or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot-leg side) completely around the U-bend to the top support ~~to the cold leg~~ of the cold-leg. Within the tubesheet this includes only the portion of the tube within the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater. The tube-to-tubesheet weld is excluded from this inspection requirement. This exclusion does not apply to tubes with sleeves installed in the tubesheet region.

9. Tube Repair refers to sleeving which is used to maintain a tube in-service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure. The following sleeve designs have been found acceptable:

   a) ABB Combustion Engineering TIG welded sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.

   b) Westinghouse laser welded sleeves, WCAP-13483, Revision 1.

10. 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 based on maintaining steam generator tube serviceability as described below:

    a) Steam generator tubes, whose degradation is 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, whose degradation is 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 4.4.5.4.a.10.c below.

SURVEILLANCE REQUIREMENTS (Continued)

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit[1] may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit[1] will be plugged or repaired.

d) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

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

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

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

(1) The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)(2)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

(2) The NDE is the value provided by the NRC in GL 95-05 as supplemented.

11. a) Bottom of the Roll Transition (BRT) - Defined as the highest point of contact between the tube and tubesheet, at or below the top of the tubesheet, as determined by eddy current testing.

    b) F* Distance - Defined as the non-degraded distance from the top of the tubesheet to the bottom of the F* length including the distance from the top of the tubesheet to the bottom of the roll transition and Non-Destructive Examination (NDE) uncertainties (i.e., F* distance = F* length + distance to BRT + NDE uncertainties).

    c) F* Length - Defined as the length of non-degraded expanded tubing below the bottom of the roll transition (BRT) or top of tubesheet, whichever is lower, that must be demonstrated to be non-degraded in order for the tube to maintain structural and leakage integrity. For the hot leg, the F* length is 1.97 inches which represents the most conservative hot leg length as defined in WCAP-16385, Revision 1. This F* length is developed from the limiting set of input parameters based on faulted conditions.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be submitted in a Special Report in accordance with 10 CFR 50.4.

b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted in a Special Report in accordance with 10 CFR 50.4 within 12 months following the completion of the inspection. This Special Report shall include:

   1. Number and extent of tubes and sleeves inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged or repaired.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

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

   1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the Commission and provide an assessment of the safety significance of the occurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No	Yes
No. of Steam Generators per Unit	Three	Three
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One[1]	One[2]

Table Notation

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instruction described in 1 above.

Provided for Information

TABLE 4.4-2
STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum Of S tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Specification 6.6.	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to Specification 6.6.	N/A	N/A

$S = \frac{9}{n}\%$ Where n is the number of steam generators inspected during an inspection.

Attachment B

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 2A-183

Proposed Technical Specification Bases Changes

Technical Specification Bases changes are provided for information only.

The following are the only affected pages:

B 3/4 4-2*

B 3/4 4-3

B 3/4 4-3a*

B 3/4 4-3b

* No changes are proposed for this page. It is provided for readability only.

3/4.4.2 (This Specification number is not used.)

3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point.

During shutdown conditions (any RCS cold leg temperature below the enable temperature specified in 3.4.9.3) RCS overpressure protection is provided by the Overpressure Protection Systems addressed in Specification 3.4.9.3.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

Safety valves similar to the pressurizer code safety valves were tested under an Electric Power Research Institute (EPRI) program to determine if the valves would operate stably under feedwater line break accident conditions. The test results indicated the need for inspection and maintenance of the safety valves to determine the potential damage that may have occurred after a safety valve has lifted and either discharged the loop seal or discharged water through the valve. Additional action statements require safety valve inspection to determine the extent of the corrective actions required to ensure the valves will be capable of performing their intended function in the future.

3/4.4.4 PRESSURIZER

The requirement that 150 kw of pressurizer heaters and their associated controls and emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate

3/4.4.5 STEAM GENERATORS (Continued)

decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural and leakage integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. However, WCAP-16385, Revision 1, "F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Beaver Valley Unit 2 Steam Generators," provides the basis for excluding that portion of the tube within the tubesheet and the tube-to-tubesheet weld, that does not contribute to structural and leakage integrity, from periodic nondestructive examination requirements. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary LEAKAGE = 150 gallons per day per steam generator). Axial cracks having a primary-to-secondary LEAKAGE less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary LEAKAGE of 150 gallons per day per steam generator can readily be detected. LEAKAGE in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in the approved vendor reports listed in Surveillance Requirement 4.4.5.4.a.9.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required of all tubes with imperfections exceeding the plugging or repair limit. Degraded steam generator tubes may be repaired by the installation of sleeves which span the degraded tube section. A steam generator tube with a sleeve installed meets the structural

3/4.4.5 STEAM GENERATORS (Continued)

requirements of tubes which are not degraded, therefore, the sleeve is considered a part of the tube. The surveillance requirements identify those sleeving methodologies approved for use. If an installed sleeve is found to have through wall penetration greater than or equal to the plugging limit, the tube must be plugged. The plugging limit for the sleeve is derived from R. G. 1.121 analysis which utilizes a 20 percent allowance for eddy current uncertainty in determining the depth of tube wall penetration and additional degradation growth. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20 percent of the original tube wall thickness.

The voltage-based repair limits of these surveillance requirements (SR) implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The guidance in GL 95-05 will not be applied to the tube-to-flow distribution baffle plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no NDE detectable cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of these SRs requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential degradation growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

3/4.4.5 STEAM GENERATORS (Continued)

where V_{Gr} represents the allowance for degradation growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

Safety analyses were performed pursuant to Generic Letter 95-05 to determine the maximum MSLB-induced primary-to-secondary leak rate that could occur without offsite doses exceeding a small fraction of 10 CFR 100 (concurrent iodine spike), 10 CFR 100 (pre-accident iodine spike), and without control room doses exceeding GDC-19. The current value of the maximum MSLB-induced leak rate and a summary of the analyses are provided in Section 15.1.5 of the UFSAR.

The mid-cycle equation in SR 4.4.5.4.a.10.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle (EOC) voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b (c) criteria.

The F* criteria incorporate the guidance provided in WCAP-16385, Revision 1, "F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Beaver Valley Unit 2 Steam Generators". F* length is the undegraded length of tubing into the tubesheet below the bottom of the roll transition (BRT) that precludes tube pullout in the event of a complete circumferential separation of the tube below the F* length. F* distance is the undegraded distance from the top of the tubesheet to the bottom of the F* length including the distance from the top of the tubesheet to the BRT and non-destructive examination (NDE) measurement uncertainties. Tubes with service-induced degradation identified in the F* distance or less than or equal to 3.0 inches below the top of the tubesheet, whichever is greater, shall be repaired or removed from service upon detection. Service induced degradation below this region in the tubesheet would be acceptable for continued operation. Therefore, based on the Farley WCAP-11306 which states that postulated leakage below the F* distance can be neglected, BVPS Unit 2 will not postulate any leakage from implementation of the F* methodology.

Tubes to which the F* inspection methodology is applied can experience through-wall degradation below the F* distance up to the limits defined in WCAP-16385, Revision 1 without increasing the probability of a tube rupture or large leakage event. Tube degradation of any type or extent below F* distance, including a complete circumferential separation of the tube, is acceptable. As applied at Beaver Valley Unit 2, the F* methodology is used to define the required tube inspection depth into the hot leg tubesheet, and is not used to permit degradation in the F* distance to remain in service.

The combined calculated leak rate from all alternate repair criteria must be less than the maximum allowable steam line break leak rate limit in any one steam generator in order to maintain doses within 10 CFR 100 guideline values and within GDC-19 values during a postulated steam line break event.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

ATTACHMENT C

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 2A-183

Westinghouse WCAP-16385-NP, March 2005
"F* Tube Plugging Criterion for Tubes with Degradation in the Tubesheet Roll Expansion Region of the Beaver Valley Unit 2 Steam Generators"

Note: The attached WCAP is Westinghouse non-proprietary.