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L-08-307

10 CFR 50.90

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

SUBJECT:
Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
License Amendment Request No. 07-007
Alloy 800 Steam Generator Tube Sleeving

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) hereby requests an amendment to the operating license for Beaver Valley Power Station (BVPS) Unit No. 2. The proposed amendment would modify Technical Specifications (TS) to allow an additional method of repair for steam generator tubes, involving use of Westinghouse leak limiting Alloy 800 sleeves. The proposed amendment would also clarify an existing reporting requirement concerning steam generator tube inspection.

The FENOC evaluation of the proposed changes is provided in Enclosure A to this letter. Additional documents that support the NRC review of this request are provided by Enclosures B through F.

Enclosures B and E contain information proprietary to Westinghouse Electric Company LLC, they are supported by affidavits signed by Westinghouse, the owner of the information. The affidavits set forth the bases on which the information may be withheld from public disclosure by the Commission and address with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Westinghouse authorization letters CAW-06-2099 and CAW-06-2100, with the accompanying affidavits, Proprietary Information Notices, and Copyright Notices, were previously provided in a letter dated September 19, 2007 (Accession No. ML072670044). FENOC requests that information 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 affidavits should reference CAW-06-2099 and/or

A001
NRR

CAW-06-2100, as appropriate, and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

This change has been reviewed by the Beaver Valley Power Station review committees. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the enclosed no significant hazard evaluation.

FENOC requests approval of the proposed amendment on or before October 12, 2009, corresponding to the start of BVPS-2 refueling outage 2R14. Implemention is planned to occur prior to achieving Mode 4 during startup from that outage.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – FENOC Fleet Licensing, at 330-761-6071.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 10, 2008.

Sincerely,

Peter P. Sena III

Enclosures:

A. FENOC Evaluation of the Proposed Change

B. WCAP-15919-P, Revision 2, "Steam Generator Tube Repair for Westinghouse Designed Plants with 7/8 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," dated January 2006 (Proprietary)

C. WCAP-15919-NP, Revision 2, "Steam Generator Tube Repair for Westinghouse Designed Plants with 7/8 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," dated January 2006 (Non-proprietary)

D. Westinghouse Letter FENOC-08-148, "Summary of Alloy 800 Sleeve Parent Tube Eddy Current Test Results," dated September 26, 2008 (Non-proprietary)

E. SG-SGDA-05-048-P, "WOG PA-MSC-0190, Revision 1: Test Results Related to TIG and Alloy 800 Sleeve Installation in 3/4 Inch and 7/8 Inch OD SG Tubing In-Service Inspection Requirements," dated January 2006 (Proprietary)

F. SG-SGDA-05-048-NP, "WOG PA-MSC-0190, Revision 1: Test Results Related to TIG and Alloy 800 Sleeve Installation in 3/4 Inch and 7/8 Inch OD SG Tubing In-Service Inspection Requirements," dated January 2006 (Non-proprietary)

cc: Mr. S. J. Collins, NRC Region I Administrator
Mr. D. L. Werkheiser, NRC Senior Resident Inspector
Ms. N. S. Morgan, NRR Project Manager
Mr. D. J. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

# Enclosure A

## FENOC Evaluation of the Proposed Changes

FENOC Evaluation of the Proposed Changes
Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 07-007

Subject: Application for amendment of Beaver Valley Power Station Unit No. 2 Technical Specifications to allow steam generator tube repair using leak limiting Alloy 800 sleeves.

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Attachments

## 1.0 SUMMARY DESCRIPTION

FirstEnergy Nuclear Operating Company (FENOC) proposes to amend Operating License NPF-73 for Beaver Valley Power Station Unit 2. The proposed changes would revise the Technical Specifications to allow the installation of leak limiting Alloy 800 sleeves as an additional approved steam generator (SG) tube repair method and would clarify an existing reporting requirement concerning SG tube inspection.

## 2.0 DETAILED DESCRIPTION

The proposed amendment would revise Technical Specification 5.5.5 to allow an additional method of repair for steam generator tubes by installation of leak limiting Alloy 800 sleeves developed by Westinghouse Electric LLC (Westinghouse) and would clarify an existing reporting requirement (Technical Specification 5.6.6.2.4) concerning SG tube inspection. The proposed method of tube repair is described in proprietary WCAP-15919-P, Revision 2, "Steam Generator Tube Repair for Westinghouse Designed Plants with 7/8 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," dated January 2006. WCAP-15919-P describes the analyses and testing performed to demonstrate that leak limiting Alloy 800 sleeves are an acceptable method of repair for steam generator tubes. Results of subsequent eddy current testing which demonstrate that parent tube degradation adjacent to the sleeve nickel band is readily detectable are documented in Reference 25, Westinghouse letter FENOC-08-148, "Summary of Alloy 800 Sleeve Parent Tube Eddy Current Test Results," dated September 26, 2008 (Non-proprietary). Test results supporting a conclusion that flaw detection capabilities within the original tube adjacent to a sleeve nickel band are not necessary in order to justify continued operation of the sleeved tube are provided by SG-SGDA-05-048-P, "WOG PA-MSC-0190, Revision 1: Test Results Related to TIG and Alloy 800 Sleeve Installation in 3/4 Inch and 7/8 Inch OD SG Tubing In-Service Inspection Requirements," dated January 2006 (Proprietary).

FENOC is proposing the following revisions.

**Technical Specification 5.5.5.2.f** provides a list of approved repair methods that are allowed by the SG program. The proposed amendment would add Specification 5.5.5.2.f.3, designating Westinghouse leak limiting Alloy 800 sleeves as described in WCAP-15919-P, Revision 2 as an additional approved repair method.

**Technical Specification 5.5.5.2.c.2** describes required SG tube plugging criteria applicable to flaws found in installed SG tube repair sleeves. Since the proposed amendment would allow installation of an additional type of repair sleeve (Westinghouse leak limiting Alloy 800 sleeves), this specification would be revised to require that the SG program contain an additional plugging criterion for the additional sleeve type. This additional criterion would require that a tube shall be plugged upon detection of any flaw in a Westinghouse leak limiting Alloy 800 sleeve. Technical Specification 5.5.5.2.c.3 already requires that a tube shall be plugged upon detection of any flaw in a sleeve to tube joint (which includes the pressure boundary of the original tube wall).

**Technical Specification 5.6.6.2.4** describes information to be reported to the NRC within 90 days after achieving Mode 4 following an outage in which the F* methodology was applied. This reporting requirement would be clarified by using terminology consistent with other reporting requirements in Specifications 5.6.6.2.1 and 5.6.6.2.2. The specification would also be clarified to explicitly state that the information to be reported involves only inspection results from the hot-leg tubesheet region (i.e., the only region where F* is permitted to be applied). This clarification is requested to prevent misinterpretation of the existing wording which implies, but does not explicitly state, that the reporting scope is limited to the region where F* was applied.

The proposed technical specification changes, which are submitted for Nuclear Regulatory Commission (NRC) review and approval, are provided in Attachment A. Deletions are shown by strike-through and insertions are shown double-underlined so that the reviewer may readily identify the information that has been deleted and added. Attachment B provides retyped pages that incorporate the changes. To meet format requirements the index and technical specifications will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

No changes are proposed for the Technical Specification Bases because the affected technical specifications are administrative and do not have associated Bases.

## 3.0 TECHNICAL EVALUATION

The proposed changes would revise the Specifications 5.5.5.2.c.2 and 5.5.5.2.f, to permit the installation of Westinghouse leak limiting Alloy 800 sleeves to repair SG tubes at Beaver Valley Power Station Unit No. 2.

### 3.1 Change to TS 5.5.5.2.f

Westinghouse provides two types of leak limiting Alloy 800 sleeves. The first type of sleeve spans the transition zone (TZ) of the original SG tube at the top of the tubesheet and is called a TZ sleeve. The upper end of a TZ sleeve is hydraulically expanded into the SG tube above the tubesheet. The lower end is hard rolled into the SG tube within the tubesheet. The length of a TZ sleeve permits the sleeve to span SG tube degradation at the top of the tubesheet. The second type of sleeve is used to span degraded areas of the SG tube at a tube support plate (TSP) elevation or in a free span section and is called a tube support (TS) sleeve. The TS sleeve is hydraulically expanded into the SG tube near each end of the sleeve. Acceptable sleeve locations for TS sleeves are from the top of the tubesheet up to the u-bend region in both the hot and cold legs.

There are two distinct advantages associated with the leak limiting Alloy 800 sleeves compared to other sleeve designs. First, no welding, brazing, or heat treatment is required during sleeve installation. Second, the strain within the tube is low, thereby reducing the likelihood of future degradation due to stress-influenced mechanisms. Although the leak limiting Alloy 800 sleeves may allow slight leakage past the sleeve (assuming the original SG tube is leaking), postulated leakage rates for normal operating and accident conditions would be extremely small compared to rates assumed in accident analyses. Any leakage would be subject to existing Technical Specification limits for operational primary to secondary leakage and accident-induced leakage, neither of which would be changed to compensate for the proposed repair method.

Steam Generator tubes with the installed sleeve meet the structural requirements of the original SG tubes and would also be subject to performance criteria for SG tube integrity required by existing Technical Specifications. Even if severance of the original SG tube is assumed in the region between the sleeve joints, the sleeve would provide the required structural support and acceptable primary to secondary leakage for normal operating and accident conditions. Testing and analysis of the sleeve and repair joint design demonstrate that these criteria are met.

In addition to the analysis and test programs discussed in WCAP-15919-P, Revision 2, a significant number of leak limiting Alloy 800 sleeves have been in operation for a number of years with no service-induced degradation. To date, no detectable leakage has been reported from tubes with Alloy 800 sleeves installed. No degradation of the installed sleeves or SG tubes in the area of the sleeve expansions has been identified.

The principal accident associated with the proposed changes is the Steam Generator Tube Rupture (SGTR) event. The consequences associated with a SGTR event are discussed in BVPS-2 Updated Final Safety Analysis Report (UFSAR) Section 15.6.3, "Steam Generator Tube Rupture." The SGTR event is a breach of the barrier between the reactor coolant system and the main steam system. The integrity of this barrier is significant from the standpoint of radiological safety in that a leaking SG tube would allow the transfer of reactor coolant into the main steam system. In the event of a SGTR, radioactivity contained in the reactor coolant would mix with water in the shell side of the affected SG. This radioactivity would be transported by steam to the turbine and then to the condenser, or directly to the condenser via the turbine bypass valves, or directly to the atmosphere via the atmospheric dump/relief valves, main steam safety valves, or the auxiliary feedwater pump turbine exhaust. Non-condensable radioactive gases in the condenser would be removed by the condenser air removal system and discharged to the plant vent. The use of leak limiting Alloy 800 sleeves would allow repair of degraded SG tubes such that the function and integrity of the SG tube is maintained. Therefore, the SGTR accident analysis results would not be affected by the use of leak limiting Alloy 800 sleeves.

If failure of a leak limiting Alloy 800 sleeve and/or the associated SG tube is assumed, the consequences of the failure would be bounded by the current SGTR analysis because the total number of plugged SG tubes (including equivalency associated with installed sleeves) is required to be consistent with accident analysis assumptions. A Main Steam Line Break (MSLB) or Feedwater Line Break (FLB) will not cause a SGTR since the sleeves are analyzed for a design basis accident differential pressure greater than that assumed in the BVPS-2 accident analyses. Tubes with sleeves would also be subject to the same safety factors as the original tubes which are described in the performance criteria for SG tube integrity in the existing Technical Specifications. These performance criteria are not being changed to compensate for the proposed repair method. The impact of sleeving on SG performance, heat transfer, and flow restriction is minimal and/or insignificant compared to plugging. The proposed BVPS-2 Technical Specification changes to allow the use of leak limiting Alloy 800 sleeves do not adversely impact any other previously evaluated design basis accident.

Evaluation of the proposed leak limiting Alloy 800 sleeves indicates no detrimental effects on the sleeve or sleeved tube assembly from reactor coolant system flow, primary coolant chemistry, secondary coolant chemistry, thermal conditions or transients, or other pressure conditions that may be experienced at BVPS-2. Postulated leakage rates for normal operating and accident conditions would be extremely small compared to rates assumed in accident analyses. Existing technical specification limits for primary to secondary operational leakage and primary to secondary accident-induced leakage would not be changed by the proposed license amendment. Data and calculation methodology concerning the reduction in primary coolant flow rate and sleeve to plug equivalency ratios are discussed in Section 10 of WCAP-15919-P, Revision 2.

The most severe loading condition assumed in WCAP-15919-P, Revision 2 bounds the corresponding BVPS-2 operating and accident value. The most severe loading on the Alloy 800 sleeve is caused by a large primary to secondary temperature difference of 127°F. This assumed loading condition is greater than the plant-specific 97°F temperature difference for BVPS-2.

WCAP-15919-P, Revision 2, describes the specific qualifications of the leak limiting Alloy 800 sleeves. The technical basis for allowing Alloy 800 sleeves to be used for tube repair is summarized below:

<u>Sleeve Installation Requirements</u>

Sleeves would be installed in accordance with processes described in WCAP-15919-P, thus ensuring consistency with credited sleeve design, qualification, installation methods, qualified examination techniques and ALARA considerations.

The parent tube in the area of the sleeve to tube hard roll joint (lower joint) and the sleeve to tube hydraulic expansion joint (upper joint) would be examined using qualified examination techniques prior to installation of the sleeve. Any service-induced indications observed in the parent tube wall at the joint location, or within 3 inches below the lower end of the lower joint of a TZ sleeve would preclude installation of the Alloy

800 sleeve. This ensures that TS 5.5.5.2.c.3 (which requires that tubes with a flaw in a sleeve to tube joint shall be plugged) would not apply. The 3 inch criterion is intended to avert the possibility that the tube would require plugging per TS 5.5.5.2.c.5.b when the F* criterion is applied.

To prepare for sleeve installation, the installation process would require the inside surface of the parent SG tube to be cleaned with a high speed buffing tool. After the parent SG tube is cleaned, the sleeve is inserted in the parent tube and positioned at the desired location. Sleeve expansion equipment is used to provide the required structural fit-up of the sleeve by forming the required number of hydraulic expansion joints with the parent tube. The expansion equipment is controlled and monitored to ensure proper diametrical expansion. For the TZ sleeve, a hard roll is performed with sleeve rolling equipment in the lower part of the sleeve to expand the sleeve into contact with the SG tube within the SG tubesheet. The torque of the rolling equipment is monitored and controlled to ensure an acceptable joint. After installation, new sleeve to tube joints undergo baseline eddy current inspection and acceptance.

Site specific procedures will be developed for the eddy current examinations of the parent tube, sleeve and sleeve joint areas. These procedures will ensure adequate coverage of the areas requiring inspection was performed.

General Structural Assessment

The Alloy 800 tubing from which the sleeves are fabricated is procured to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section II, Part B, SB-163, NiFeCr Alloy, Unified Numbering System N08800, and Section III, Subsection NB-2000. Alloy 800, incorporated in ASME Code Case N-20 and subsequently into a later edition of the code, is considered acceptable for use by Regulatory Guide 1.85, "Materials Code Case Acceptability ASME Section III, Division 1," Revision 24, dated July 1986. Additionally, supplemental requirements are imposed to more tightly control the parameters within the limits allowed by the ASME specification.

Analyses of the sleeved tube assemblies have been completed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976.

SG tubes with installed leak limiting Alloy 800 sleeves meet the structural integrity requirements of the original SG tube. Even if severance of the parent SG tube is assumed in the region between the sleeve joints, the sleeve would provide the required structural support and acceptable primary to secondary leakage for normal operating and accident conditions. Testing and analysis of the sleeve and repair joint design demonstrate that these criteria are met.

The Alloy 800 sleeve material showed no signs of degradation under high temperature and pressure conditions in a caustic environment, while sleeve/tube specimens maintained primary side pressure and exhibited no leakage throughout the duration of the test program. Earlier design variations of this sleeve/tube assembly (larger diametrical hydraulic expansion or varying number of expansions/configurations) were used at KORI 1 (South Korea) and Tihange 3 (Belgium) steam generators. The current design configuration is in service or recently in service at Angra 1 (Brazil), KRSKO (Slovenia), Ringhals 4 (Sweden), Tihange 2 (Belgium), Ulchin 1 & 2 (South Korea), Calvert Cliffs 1 and 2, Comanche Peak 1, and Watts Bar 1.

Section 8.2 of WCAP-15919-P describes how RG 1.121 was used to develop the structural limit of the sleeve should sleeve wall degradation occur. Electric Power Research Institute (EPRI) Technical Report 10001191, "Steam Generator Degradation Specific Management Flaw Handbook," dated 2001, was used as the basis for adding margin to account for the configuration of a long axial crack. Leak limiting Alloy 800 sleeves are shown (by test and analysis) to retain burst strength in excess of three times the normal operating pressure differential at end of operating cycle conditions. No credit for the presence of the parent SG tube between the sleeve joints was assumed in the minimum wall burst evaluation for the leak limiting Alloy 800 sleeve. Bounding normal operating, design seismic, and transient loading conditions on the SG tube sleeves were used for the structural analysis of the sleeves and are summarized in Section 8.5 of WCAP-15919-P. The loading conditions assumed in Section 8.5 of WCAP-15919-P (supplemented by Westinghouse calculation note CN-SGDA-04-4) are at least as conservative as the reactor coolant system design transients described in BVPS-2 FSAR Table 3.9N-1.

Corrosion Assessment

A significant number of Alloy 800 leak limiting sleeves have been in operation for a number of years with no service-induced degradation. Additionally, no detectable leakage has been associated with a tube that contains an Alloy 800 leak limiting sleeve. Accelerated corrosion testing of leak limiting Alloy 800 sleeve/tube assemblies has been performed in simulated primary and secondary side SG environments and the leak limiting Alloy 800 sleeves showed no signs of cracking in either the primary or secondary side tests. Leak limiting Alloy 800 sleeve corrosion performance is discussed in Section 6 of WCAP-15919-P, Revision 2.

Corrosion testing of sleeve/tube assemblies has been performed in Belgium (Laborelec Laboratories) and the U.S. (Westinghouse) with satisfactory results. These results, when analyzed in conjunction with corrosion test results from the tungsten inert gas-welded sleeve program, confirm the adequacy of the sleeve joint design.

Mechanical Integrity Assessment

Mechanical testing of leak limiting Alloy 800 sleeve/tube assemblies was performed using mock SG tubes. The tests determined axial load, collapse pressure, burst pressure, leak rates, wear, and thermal cycling capability. The demonstrated load capacity of the assemblies provides an adequate safety factor for normal operating and postulated accident conditions.

The load capacity of the upper and lower sleeve joints is sufficient to withstand thermally induced stresses and displacements resulting from the temperature differential between the sleeve and the SG tube and pressure-induced stresses resulting from normal operating and accident conditions. The burst and collapse pressures of the sleeve provide margin over the limiting pressure differential. The mechanical testing demonstrated that the installed sleeve will withstand the cyclic loading resulting from power changes in the plant and other transients. The loading conditions developed in Section 8 of WCAP-15919-P were used to develop the conditions for the mechanical tests described in Section 7 of WCAP-15919-P. The temperature and pressure differentials described in Section 8 of WCAP-15919-P are conservative with respect to BVPS-2 operating and accident conditions.

Leak Rate Assessment

Section 7 of WCAP-15919-P provides details regarding how conservative normal operating and accident leak rate values were determined and provides the basis for leak rate values used for calculations such as condition monitoring or operational assessments. The Alloy 800 TZ and TS sleeve leakage characteristics were evaluated at shutdown, normal operating, and accident temperatures so that all possible plant conditions would be enveloped by the test results.

Technical specifications reflect acceptable limits for primary to secondary leakage assumed in accident analyses. Leakage from SG tube sleeves would be extremely small compared to rates assumed in accident analyses and permitted by technical specifications. Therefore, adequate leakage margin exists to accommodate a sufficient number of sleeves to satisfy foreseeable needs while continuing to satisfy existing limits for primary to secondary operational leakage and primary to secondary accident-induced leakage.

Leakage calculations such as for condition monitoring or operational assessments would conservatively assume that each installed sleeve leaks at the upper 95 percent confidence limit on the mean value of leakage for appropriate temperature and pressure conditions. The total projected leak rate from all sleeves would be combined with the total amount of leakage from all other sources (e.g., Alternate Repair Criteria (ARC), F* methodology and non-alternate repair criteria indications) for comparison against the limits on accident-induced leakage.

Pre-installation Inspection of the Lower Hard Rolled Joint

Since qualification of eddy current techniques consistent with the requirements of Reference 9 has not been performed for that portion of the parent tube that is in contact with (adjacent to) the nickel band of the Alloy 800 sleeve and forms part of the lower pressure boundary of the TZ sleeved joint, this area of the parent tube will be examined prior to installation of the sleeve with a Plus Point coil or equivalent using examination techniques qualified per Appendix H of Reference 9. Any service-induced indications observed in this region would preclude installation of the Alloy 800 sleeve. Since the sleeve to tube hard roll joint isolates the parent tube from the initiating environment necessary for Primary Water Stress Corrosion Cracking (PWSCC), the potential for SCC initiation is effectively eliminated. Additionally, for BVPS-2, the potential for PWSCC development within the expanded tube in tubesheet region is reduced due to the application of shot peening prior to commercial operation. For several outages FENOC has performed Plus Point inspection of bugle and over-expansion signals in the expanded tube portion of the tubesheet at BVPS-2. To date no degradation has been reported.

Post-installation Inspections:

As required by EPRI Technical Report 1013706 (previously identified as TR 1003138 R6), post-installation (pre-service) examination will be performed on the full length of 100 percent of the leak limiting Alloy 800 sleeve/tube assemblies using a Plus Point probe or equivalent. This examination will establish in-service inspection baseline data and initial installation acceptance data on the primary pressure boundary of the sleeve/tube assembly repair.

In-service Inspection:

In-service inspection of the sleeved tubes would be done as part of the periodic inspection program of the steam generator tubing using eddy current testing techniques. In-service inspection would be performed using a Plus Point probe or equivalent. Other coils and/or methods would be considered for any complementary inspection capability they may provide. The inspection method has been used in operating steam generators in fourteen plants for both the initial installation acceptance and the subsequent in-service inspection.

An equivalency assessment was performed to establish that the essential variables developed for eddy current inspection of 3/4 inch tube sleeves can also be applied to 7/8 inch tube sleeves. The eddy current inspection method to be used has a documented qualification, per Appendix H of EPRI Technical Report TR-107569-V1R5, "PWR Steam Generator Examination Guidelines: Revision 5, Volume 1: Requirements," dated September 1997. This qualification addressed degradation of the parent tube at the upper joint region and on the sleeve. Degradation of the parent tube adjacent to the nickel band region was not considered a relevant flaw location at the time of the qualification. Since a Plus Point (or equivalent) inspection of the parent tube will be performed in the hardroll joint region prior to sleeve installation, subsequent degradation of the parent tube in this area after sleeve installation is not expected to occur. However, testing described in Reference 25 was performed to demonstrate that parent tube degradation adjacent to the sleeve nickel band is readily detectable by eddy current inspection.

For sleeving inspections, detection capability would be established with an operational margin relative to structurally limiting flaws. Flaws would be plugged upon detection.

Accordingly, it would not be necessary to determine flaw sizes so that flaws could remain in service. By this approach, sizing accuracy does not need to be quantified.

Effects of Sleeving on Operation

The effects of sleeve installation on SG heat removal capability and reactor coolant system flow rate are discussed in Section 10 of WCAP-15919-P, which states that the installation of the sleeves does not substantially affect the primary system flow rate or the heat transfer capability of the SGs. The typical hydraulic equivalency of plugs and installed sleeves, called the sleeve/plug ratio or sleeve to plug equivalency ratio, is contained in Table 10-1 of WCAP-15919-P for different configurations of TZ sleeves and TS sleeves in 7/8 inch OD SG tubes. The Table 10-1 sleeve/plug ratio values are an approximation only based on assumed operating parameters and sleeve types for SGs with 7/8 inch tubes, and some variations in the sleeve/plug ratio will occur based on operating parameters and SG conditions. FENOC will use the sleeve/plug ratio values contained in Table 10-1 of WCAP-15919-P to determine the equivalent SG plugging due to installed leak limiting Alloy 800 sleeves, unless more appropriate values become available. The total SG plugging level for each SG will be determined by adding the equivalent SG plugging percentage due to installed leak limiting Alloy 800 sleeves to the percent of SG tubes plugged. The total number of plugged SG tubes (including equivalency associated with installed sleeves) is required to be consistent with accident analysis assumptions.

Alloy 800 was designed for SG tubing as an alternative to Alloy 600 and is comprised of the same three major metallurgical components (nickel, iron, chrome) as Alloys 600 and 690. It has been in use in SGs for many years in European nuclear plants and has performed well in a primary chemistry environment similar to BVPS-2. Therefore, Alloy 800 is compatible with the primary chemistry regime used at BVPS-2 and no changes to this regime are necessary.

Relationship between TS Sleeves and TSP ARC

For sleeves installed at Tube Support Plate (TSP) locations, TSP Outside Diameter Stress Corrosion Cracking (ODSCC) ARC will not apply to the sleeved TSP intersections. No Technical Specification change is necessary to clarify this relationship for sleeved tubes because the sleeve will remove the TSP ODSCC ARC indications from service.

Severe Accident Considerations

Severe accidents can lead to high primary pressure of 2500 psi and high primary temperature between 1200°F and 1500°F. At severe accident conditions, pressure tends to loosen the tube joint and temperature tends to tighten it. As the temperature reaches 1500°F, both the sleeve and tube yield at steam line break pressures.

Because the sleeve material is specified to have a low yield stress (30 ksi minimum and controlled maximum), the sleeve will yield at a lower temperature (or pressure) than the tube, thereby tending to tighten the tube joint. At 1500°F, the ultimate stress of the sleeve material is comparable to that of the SG tube and the integrity of the sleeve repair is commensurate with the integrity of the inservice SG tubes. Therefore, under severe accident conditions, sleeving is expected to have no impact on plant performance.

Inspection of Parent Tube Adjacent to Nickel Band

Regarding parent tube flaw detection capabilities adjacent to the nickel band, references 19 and 20 discuss detectability of electro-discharge machine (EDM) notches applied to the parent tube. Reference 20 shows that parent tube degradation of at least 70 percent through-wall would be readily detectable based on observations of flaw amplitude response for EDM flaws. Reference 25 describes test results using the Plus Point probe to examine sleeved tube assemblies containing laboratory generated ODSCC flaws. Four axially oriented crack networks were located at 90 degree spacing around the tube circumference. Three of the four crack networks included 100 percent through-wall degradation. One had a local maximum of 85 percent through-wall. The Plus Point 300 kHz flaw amplitudes for the parent tube flaws after expansion into a tubesheet collar and prior to sleeve installation were 7.5 to 1.3 volts. These (300 kHz) flaw voltages are approximately equal to PWSCC depths of 100 to 60 percent through-wall for equivalent flaw amplitude responses. The results of this program show that for the currently qualified detection frequencies of 75 and 150 kHz, that the Plus Point coil detected the four individual flaws used for the test. Comparison of flaw signal responses for EDM axial flaw simulations for locations adjacent to and not adjacent to the nickel band region show that the influence upon flaw amplitude response due to the nickel band is limited, and that part through-wall parent tube degradation would be readily detectable adjacent to the nickel band. Test results described in Reference 25 show that parent tube degradation of modest depths would be readily detectable through the sleeve, adjacent to the nickel band, using the Plus Point coil.

TZ Sleeve Lower Joint Integrity with Flawed Parent Tube

Prior to establishing that flaws in the parent tube adjacent to the nickel band can be reliably detected, a testing program described in SG-SGDA-05-048 was conducted as technical basis that degradation of the parent tube adjacent to the nickel band would not impair tube integrity irrespective of the size of the flaws. The testing confirmed that undetected axial and circumferential degradation of the parent tube adjacent to the sleeve nickel band, regardless of depth, would not prevent the sleeve from satisfying its design requirements. Thus, flaw detection capabilities within the parent tube adjacent to the sleeve nickel band are not necessary in order to justify continued operation of the sleeved tube. San Onofre Nuclear Generating Stations, Unit Nos. 2 and 3, have been granted relief from inspection requirements in the nickel band region of a sleeve joint on this basis (Reference 17). Nevertheless, present day eddy current inspection methods can adequately detect part through-wall degradation of the parent tube inside diameter. Based on this detection capability, sleeved tubes with degradation of the parent tube in this region would be removed from service by plugging prior to the degradation of the parent tube achieving the depths considered in the SG-SGDA-05-048 test program.

Inward Deformation of Sleeves

The TIG sleeve and (to a much lesser extent) the laser welded sleeve have experienced inward deformations of the sleeve due to entrapped water between the tube and sleeve. These deformations affected approximately one half of the sleeve inside diameter and in no cases was the sleeve primary flow area completely closed. Leakage past either the upper or lower joints, or through the parent tube degradation which necessitated the repair is expected to be the cause. Both of these sleeve designs utilize a welded upper joint. As the Alloy 800 sleeve upper joint is a leak-limiting joint, inward deformation of the sleeve due to entrapped water is not a relevant issue. Section 7.2.2 and Table 7-1 of WCAP-15919 describes collapse testing which demonstrated that fluid entrapped between the sleeve and the parent tube "vents" before pressure is great enough to collapse the sleeve. Additionally, a leak allowance per sleeve is accounted for during the operational assessment process.

Variability in Coefficient of Thermal Expansion

NRC review of license amendment requests for a tubesheet region alternate repair criterion for plants with hydraulically expanded tube-in-tubesheet joint configurations has resulted in questions regarding the applicability of ASME Code coefficient of thermal expansion data. On-going testing under this alternate repair criterion program has shown that the application of ASME Code values is appropriate. Therefore, SG-SGDA-05-048 should provide a conservative representation of joint integrity. The industry will be responding to NRC on this subject under the alternate repair criterion program.

3.2 Change to TS 5.5.5.2.c.2

The RG 1.121 bounding structural limit for leak limiting Alloy 800 sleeves is 45 percent through-wall detailed in Section 8.2.1 of WCAP-15919-P, Revision 2, which is based on normal operating conditions for the worst case envelopment of SG conditions for Westinghouse Model 44, 44F, and 51 steam generators. However, the proposed TS criterion would require any tube with a leak limiting Alloy 800 sleeve would be plugged upon detection of a flaw in the sleeve. Existing TS 5.5.5.2.c.3 requires plugging of tubes with a flaw in a sleeve to tube joint.

To ensure that a defect in the pressure boundary of a sleeve does not adversely impact the leakage integrity of the sleeve, a sleeved tube will be plugged if a flaw is detected in any portion of the sleeve. In addition, to ensure that a defect in the pressure boundary of a sleeve does not adversely impact the structural integrity of the sleeve, a sleeved tube will be plugged if a flaw is detected in the parent tube wall in the sleeve to tube joint.

Flaws that are detected in the parent tube between the sleeve to tube joints do not impact the pressure boundary of the sleeve/tube assembly and do not impact the structural integrity of the sleeve. Therefore, plugging of the sleeved tube is not required for degradation in non-pressure boundary locations. Reference 1 describes the portions of the parent tube which are classified as pressure boundary and non-pressure boundary.

3.3 Changes to TS 5.6.6.2.4

Proposed changes to Technical Specification 5.6.6.2.4 involve reporting requirements for information reported to the NRC within 90 days after achieving Mode 4 following an outage in which the F* methodology was applied. Other reporting requirements in Specifications 5.6.6.2.1 and 5.6.6.2.2 use the phrase, "A report shall be submitted within ... days after the initial entry into MODE 4 following ... ." The use of terminology similar to Specifications 5.6.6.2.1 and 5.6.6.2.2 would reinforce the original intent of the reporting requirement, would provide consistency among various reporting requirements, and would remove any potential for misinterpretation.

Technical Specification 5.6.6.2.4 would also be clarified to explicitly state that the information to be reported involves only inspection results from the hot-leg tubesheet region (i.e., the only region where F* is permitted to be applied). The existing reporting is required to permit the NRC staff to verify that operating experience continues to be conservative relative to the assumptions made in License Amendment No. 160 (i.e., the F* amendment). Therefore, the information that is required to be reported would be limited in scope to portions of tubes affected by F*. This clarification is requested to preserve the intent of the reporting requirement by explicitly stating that the reporting scope is limited to the region where F* was applied.

4.0 REGULATORY EVALUATION

FirstEnergy Nuclear Operating Company proposes to revise Beaver Valley Power Station Unit No. 2 (BVPS-2) Technical Specification 5.5.5 to allow an additional method of repair for steam generator (SG) tubes through the use of leak limiting Alloy 800 sleeves developed by Westinghouse Electric LLC (Westinghouse). An additional revision would clarify an existing reporting requirement concerning SG tube inspection in Technical Specification 5.6.6.2.4. Specifically, the proposed technical specification changes are described as follows:

**Technical Specification 5.5.5.2.f** provides a list of approved repair methods that may be allowed by the SG program. The proposed amendment would add Specification 5.5.5.2.f.3, designating Westinghouse leak limiting Alloy 800 sleeves as described in WCAP-15919-P, Revision 2 as an additional approved repair method.

**Technical Specification 5.5.5.2.c.2** describes required SG tube plugging criteria applicable to flaws found in installed SG tube repair sleeves. Since the proposed amendment would allow installation of an additional type of repair sleeve (i.e., Westinghouse leak limiting Alloy 800 sleeves), this specification would be revised to require that the SG program contain an additional criterion for the additional sleeve type. This additional criterion would require that a tube shall be plugged upon detection of any flaw in a Westinghouse leak limiting Alloy 800 sleeve.

**Technical Specification 5.6.6.2.4** describes information to be reported to the NRC within 90 days after achieving Mode 4 following an outage in which the F* methodology was applied. This reporting requirement would be clarified by using terminology consistent with other reporting requirements in Specifications 5.6.6.2.1 and 5.6.6.2.2. The specification would also be clarified to explicitly state that the information to be reported involves only inspection results from the hot-leg tubesheet region (i.e., the only region where F* is permitted to be applied).

4.1 Significant Hazards Consideration

FirstEnergy Nuclear Operating Company (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?**

   **Response:** No. The leak limiting Alloy 800 sleeves are designed using the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and, therefore, meet the design objectives of the original steam generator (SG) tubing. The applied stresses and fatigue usage for the sleeves are bounded by the limits established in the ASME Code. Mechanical testing has shown that the structural strength of sleeves under normal, upset, emergency, and faulted conditions provides margin to the acceptance limits. These acceptance limits bound the most limiting (three times normal operating pressure differential) burst margin recommended by NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Burst testing of sleeve/tube assemblies has confirmed the analytical results and demonstrated that no unacceptable levels of primary to secondary leakage are expected during any plant condition.

   The leak limiting Alloy 800 sleeve depth-based structural limit is determined using NRC guidance and the pressure stress equation of ASME Code, Section III with additional margin added to account for the configuration of long axial cracks. Calculations show that a depth-based limit of 45 percent through-wall degradation is acceptable. However, the proposed amendment provides additional margin by requiring an Alloy 800 sleeved tube to be plugged on detection of any flaw in the sleeve or in the pressure boundary portion of the original tube wall in the sleeve to tube joint. Degradation of the original tube adjacent to the nickel band of an Alloy 800 sleeve, regardless of depth, would not prevent the sleeve from satisfying design requirements. Thus, flaw detection capabilities within the original tube adjacent to the sleeve nickel band are not necessary in order to justify continued operation of the sleeved tube.

Evaluation of repaired SG tube testing and analysis indicates no detrimental effects on the leak limiting Alloy 800 sleeve or sleeved tube assembly from reactor system flow, primary or secondary coolant chemistries, thermal conditions or transients, or pressure conditions as may be experienced at BVPS-2. Corrosion testing and historical performance of sleeve/tube assemblies indicates no evidence of sleeve or tube corrosion considered detrimental under anticipated service conditions.

Implementation of the proposed change has no significant effect on either the configuration of the plant or the manner in which it is operated. The consequences of a hypothetical failure of the leak limiting Alloy 800 sleeve/tube assembly are bounded by the current SG tube rupture (SGTR) analysis described in the BVPS-2 Updated Final Safety Analysis Report because the total number of plugged SG tubes (including equivalency associated with installed sleeves) is required to be consistent with accident analysis assumptions. A main steam line break or feedwater line break would not cause a SGTR since the sleeves are analyzed for a maximum accident differential pressure greater than that predicted in the BVPS-2 safety analysis. The sleeve/tube assembly leakage during plant operation would be minimal and is well within the allowable Technical Specification leakage limits and accident analysis assumptions, neither of which would be changed to compensate for the proposed repair method.

Proposed changes to Technical Specification 5.6.6.2.4 only affect a reporting requirement and do not affect plant design, operation or maintenance. They are intended as clarifications that would reinforce the original intent of the reporting requirement.

Therefore, the proposed change does 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?**

   **Response:** No. The leak limiting Alloy 800 sleeves are designed using the applicable ASME Code as guidance, and therefore meet the objectives of the original SG tubing. As a result, the functions of the SG will not be significantly affected by the installation of the proposed sleeve. Therefore, the only credible failure mode for the sleeve and/or tube is to rupture, which has already been evaluated. The continued integrity of the installed sleeve/tube assembly is periodically verified as required by the Technical Specifications and a sleeved tube will be plugged on detection of a flaw in the sleeve or in the pressure boundary portion of the original tube wall in the sleeve to tube joint.

Proposed changes to Technical Specification 5.6.6.2.4 only affect a reporting requirement and do not affect plant design, operation or maintenance. They are editorial in nature and are intended as clarifications that would reinforce the original intent of the reporting requirement.

Implementation of the proposed change has no significant effect on either the configuration of the plant, or the manner in which it is operated.

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?**

**Response:** No. The repair of degraded SG tubes with leak limiting Alloy 800 sleeves restores the structural integrity of the degraded tube under normal operating and postulated accident conditions. The reduction in reactor coolant system flow due to the addition of Alloy 800 sleeves is not significant because the cumulative effect of all repaired (sleeved) and plugged tubes will continue to allow reactor coolant flow to be greater than the flow limit established in the Technical Specification LCO 3.4.1. The design safety factors utilized for the sleeves are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in the original SG design. Tubes with sleeves would also be subject to the same safety factors as the original tubes which are described in the performance criteria for SG tube integrity in the existing Technical Specifications. These performance criteria are not being changed to compensate for the proposed repair method. The sleeve and portions of the installed sleeve/tube assembly that represent the reactor coolant pressure boundary will be monitored and a sleeved tube will be plugged on detection of a flaw in the sleeve or in the pressure boundary portion of the original tube wall in the leak limiting sleeve/tube assembly. Use of the previously identified design criteria and design verification testing ensures that the margin of safety is not significantly different from the original SG tubes.

Proposed changes to Technical Specification 5.6.6.2.4 only affect a reporting requirement and do not affect plant design, operation or maintenance. They are editorial in nature and are intended as clarifications that would reinforce the original intent of the reporting requirement.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, FENOC concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.2 Applicable Regulatory Requirements/Criteria

Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50), Appendix A, "General Design Criteria for Nuclear Power Plants" Criterion 14, Reactor Coolant Pressure Boundary, contains requirements applicable to SG tubes since they are part of the reactor coolant pressure boundary. Criterion 14 requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross failure. The repair of the existing reactor coolant pressure boundary is performed in accordance with Section XI of ASME Boiler and Pressure Vessel Code, which refers to Section III of the ASME Code. The original SG tubes were designed in accordance with Section III of the ASME Code (1971 edition through Summer 1972 Addenda). The design criteria for the leak limiting sleeves were established to meet the loading condition and stress requirements of Section III of the ASME Code (1995 edition, no addenda), which is consistent with the section of the ASME Code that applies to the original SG tubes.

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

Regulatory Guide (RG) 1.121 provides guidance for determining the minimum wall thickness at which a SG tube should be plugged. The RG 1.121 performance criteria recommend that the margin of safety against SGTR under normal operating conditions should not be less than three at any tube location where defects have been detected. The margin of safety against tube failures under postulated accident conditions should be consistent with the margin of safety determined by the stress limits specified in the ASME Code. The RG 1.121 requirements were used to develop the structural limit of leak limiting

Alloy 800 sleeves should sleeve wall degradation occur. In addition, the fatigue and stress analysis of the sleeved tube assemblies have been completed in accordance with the requirements of RG 1.121.

Based on past usage, extensive testing and analysis, the leak limiting Alloy 800 sleeves provide satisfactory repair of degraded SG tubes. Qualified nondestructive examination techniques will be used to perform necessary sleeve and original SG tube inspections for defect detection, and to verify proper installation of the repair sleeve.

4.3 Precedent

The following table identifies license amendments that have been granted to allow repair of steam generator tubes using Alloy 800 sleeves. References are provided in Section 6.0 of this LAR.

| Plant | Amendment No. | Date | Topical Report | Reference No. |
|---|---|---|---|---|
| Calvert Cliffs 1 | 231 | 09/01/1999 | CEN-633-P Rev. 3 | 12 |
| Calvert Cliffs 2 | 207 | 09/01/1999 | CEN-633-P Rev. 3 | 12 |
| Watts Bar 1 | 44 | 08/15/2003 | WCAP-15918-P Rev. 0 | 13 |
| Comanche Peak 1 | 112 | 03/24/2004 | WCAP-15918-P Rev. 1 | 14 |
| Beaver Valley 1 | 260 | 10/05/2004 | WCAP-15919-P Rev. 0 | 15 |
| St. Lucie 2 | 144 | 04/18/2006 | WCAP-15918-P Rev. 2 | 16 |

Southern California Edison has also been granted license amendments (Reference17) for San Onofre Nuclear Generating Stations, Unit Nos. 2 and 3, that provide relief from inspection requirements in the nickel band region of a sleeve joint for a limited time period based on demonstration that the structural and leakage integrity of the sleeve-to-tube joint would be maintained without taking any credit for the portion of the joint that was exempted from inspection. FENOC is proposing to inspect the parent tube adjacent to the nickel band so that similar relief would not be necessary for BVPS-2.

4.4 Conclusions

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.

## 5.0 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.

## 6.0 REFERENCES

1. Westinghouse Electric LLC WCAP-15919-P, Revision 2, "Steam Generator Tube Repair for Westinghouse Designed Plants with 7/8 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," dated January 2006 (proprietary).

2. Westinghouse Electric LLC WCAP-15919-NP, Revision 2, "Steam Generator Tube Repair for Westinghouse Designed Plants with 7/8 Inch Inconel 600 Tubes Using Leak Limiting Alloy 800 Sleeves," dated January 2006 (non-proprietary).

3. Westinghouse Electric LLC SG-SGDA-05-048-P, "WOG PA-MSC-0190, Revision 1: Test Results Related to TIG and Alloy 800 Sleeve Installation in 3/4 Inch and 7/8 Inch OD SG Tubing In-Service Inspection Requirements," dated January 2006 (proprietary).

4. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section II, Part B, SB-163, NiFeCr Alloy UNS N08800, and Section III, Subsection NB-2000.

5. Regulatory Guide 1.85, "Materials Code Case Acceptability ASME Section III, Division 1," Revision 24, dated July 1986.

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

7. EPRI Technical Report 10001191, "Steam Generator Degradation Specific Management Flaw Handbook," dated 2001.

8. EPRI Technical Report TR-107569-V1R5, "PWR Steam Generator Examination Guidelines: Revision 5, Volume 1: Requirements," dated September 1997.

9. EPRI Technical Report 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6, Requirements," dated October 2002.

10. EPRI Technical Report 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 7, Requirements," dated October 2007.

11. Westinghouse Electric LLC WCAP-15918-P, Revision 00, (CEN-630-P, Revision 05-P), "Steam Generator Tube Repair for Combustion Engineering and Westinghouse Designed Plants with 3/4 Inch Inconel 600 tubes Using Leak Limiting Alloy 800 Sleeves," dated November 2002 (proprietary).

12. License Amendment No. 231 to Facility Operating License No. DPR-53 and Amendment No. 207 to Facility Operating License No. DPR-69, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Issuance of Amendment RE: Steam Generator Tube Repair Using Leak Limiting Alloy 800 Sleeves (TAC Nos. MA4278 and MA4279)," dated September 1, 1999 (ADAMS # ML010520103).

13. License Amendment No. 44 to Facility Operating License No. NPF-90, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment For Steam Generator Tube Repair (TAC No. MB6976)," dated August 15, 2003 (ADAMS # ML 032300143).

14. License Amendment No. 112 to Facility Operating License No. NPF-87, "Comanche Peak Steam Electric Station (CPSES), Unit 1 - Issuance of Amendment Re: Steam Generator Tube Repair Using Leak Limiting Alloy 800 Sleeves (TAC No. MC0197)," dated March 24, 2004 (ADAMS # ML040840401).

15. License Amendment No. 260 to Facility Operating License No. DPR-66, "Beaver Valley Power Station, Unit No. 1 - Issuance of Amendment Re: One-Cycle Use of Westinghouse Leak-Limiting Alloy 800 Steam Generator Tube Sleeves For SG Tube Repair (TAC No. MC1857)," dated October 5, 2004 (ADAMS # ML042400116).

16. License Amendment No. 144 to Facility Operating License No. NPF-16, "St. Lucie Plant, Unit No. 2 - Issuance of Amendment Regarding Use of Westinghouse Alloy 800 Sleeves in Steam Generators (TAC No. MC5633)," dated April 18, 2006 (ADAMS # ML061070115).

17. License Amendment Nos. 215 and 207 to Facility Operating License Nos. NPF-10 and NPF-15, "San Onofre Nuclear Generating Station, Units 2 and 3 - Issuance of Amendments Re: Steam Generator Tube Surveillance Program, Tube Repair (TAC NOS. MD2584 AND MD2585)," dated November 29, 2007 (ADAMS # ML073170529).

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

19. Westinghouse Electric LLC, WOG-04-518, "Engineering Position Paper on NDE Issues Related to TIG and Alloy 800 Sleeves with Regard to Sleeve Nickel Band NRC Discussion," October 11, 2004.

20. Westinghouse Electric LLC, WOG-05-338, "Engineering Position Paper on NDE Issues Related to TIG and Alloy 800 Sleeves with Regard to Sleeve Nickel Band NRC Discussion, Revision 1," July 19, 2005.

21. Westinghouse Electric LLC, WCAP-13698, Revision 3, "Laser Welded Sleeves for ¾ Inch Diameter Tube Steam Generators Generic Sleeving Report," July 1998.

22. USNRC, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," Facility Operating License No. NPF-87 TXU Generation Company LP Comanche Peak Steam Electric Station, Unit 1 Docket No. 50-445, dated December 13, 2005.

23. USNRC, NUREG/CR-5752, "Assessment of Current Understanding of Mechanisms of Initiation, Arrest, and Reinitiation of Stress Corrosion Cracks in PWR Steam Generator Tubing," February 2000.

24. Combustion Engineering, INC., CEN-617-P, "Steam Generator Tube Repair for Tubes Containing Westinghouse Mechanical Sleeve Using Leak Limiting I690 Sleeves," March 1995.

25. Westinghouse Letter FENOC-08-148, "Summary of Alloy 800 Sleeve Parent Tube Eddy Current Test Results," dated September 26, 2008 (Non-proprietary).

## ATTACHMENT A

**Beaver Valley Power Station, Unit No. 2**
**License Amendment Request No. 07-007**

**Proposed Technical Specification Changes**

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The following are the only affected pages:

5.5-6*

5.5-7*

5.5-8

5.5-9*

5.5-10*

5.5-11

5.6-5*

5.6-6

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

5.5 Programs and Manuals

5.5.5.1 Unit 1 Steam Generator (SG) Program (continued)

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. During each period inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three intervals between refueling outages (whichever is less) without being inspected.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE

5.5.5.2 Unit 2 Steam Generator Program

a. Provisions for Condition Monitoring Assessments

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

b. Provisions for Performance Criteria for SG Tube Integrity

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

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes

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

retaining a safety factor of 3.0 against burst under normal steady state full power operation primary to secondary pressure differential and, except for flaws addressed through application of the alternate repair criteria discussed in Specification 5.5.5.2.c.4, a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

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

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Except during a SG tube rupture, leakage from all sources excluding the leakage attributed to the degradation described in Specification 5.5.5.2.c.4 is also not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, “RCS Operational Leakage.”

c. Provisions for SG Tube Repair Criteria

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

5.5 Programs and Manuals

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

where:

$V_{URL}$ = upper voltage repair limit

$V_{LRL}$ = lower voltage repair limit

$V_{MURL}$ = mid-cycle upper voltage repair limit based on time into cycle

$V_{MLRL}$ = mid-cycle lower voltage repair limit based on $V_{MURL}$ and time into cycle

$\Delta t$ = 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). The NDE is the value provided by the NRC in GL 95-05 as supplemented.

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

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

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

   a) Tubes with no portion of a lower sleeve joint in the hot-leg tubesheet region shall be repaired or plugged upon detection of any flaw identified within 3.0 inches below the top of the tubesheet or within 2.2 inches below the bottom of roll transition, whichever elevation is lower. Flaws located below this elevation may remain in service regardless of size.

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

d. Provisions for SG Tube Inspections

Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. In addition to meeting the requirements of d.1, d.2, d.3, d.4, and d.5 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

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

5.5 Programs and Manuals

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

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

4. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (Specification 5.5.5.2.c.4) shall be inspected by bobbin coil probe during all future refueling outages.

   Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

5. When the F* methodology has been implemented, inspect 100% of the inservice tubes in the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of Specification 5.5.5.2.c.5 every 24 effective full power months or one interval between refueling outages (whichever is less).

e. Provisions for monitoring operational primary to secondary LEAKAGE

f. Provisions for SG Tube Repair Methods

   Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

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

   2. Westinghouse laser welded sleeves, WCAP-13483, Revision 2.

   3. <u>Westinghouse leak-limiting Alloy 800 sleeves, WCAP-15919-P, Revision 2.</u>

5.6 Reporting Requirements

5.6.6 Steam Generator Tube Inspection Report (continued)

5.6.6.2 Unit 2 SG Tube Inspection Report

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

    a. The scope of inspections performed on each SG,

    b. Active degradation mechanisms found,

    c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

    f. Total number and percentage of tubes plugged or repaired to date,

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

    h. The effective plugging percentage for all plugging and tube repairs in each SG, and

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

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

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

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

5.6 Reporting Requirements

5.6.6.2 Unit 2 Steam Generator Tube Inspection Report (continued)

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

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

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

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

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

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

## ATTACHMENT B

**Beaver Valley Power Station, Unit No. 2**
**License Amendment Request No. 07-007**

**Retyped Technical Specification Pages**

The following are the only affected pages:

5.5-8

5.5-11

5.6-6

5.5 Programs and Manuals

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

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

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

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

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

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

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

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

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

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

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

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

4. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (Specification 5.5.5.2.c.4) shall be inspected by bobbin coil probe during all future refueling outages.

   Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

5. When the F* methodology has been implemented, inspect 100% of the inservice tubes in the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of Specification 5.5.5.2.c.5 every 24 effective full power months or one interval between refueling outages (whichever is less).

e. Provisions for monitoring operational primary to secondary LEAKAGE

f. Provisions for SG Tube Repair Methods

   Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

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

   2. Westinghouse laser welded sleeves, WCAP-13483, Revision 2.

   3. Westinghouse leak-limiting Alloy 800 sleeves, WCAP-15919-P, Revision 2.

5.6 Reporting Requirements

5.6.6.2 Unit 2 Steam Generator Tube Inspection Report (continued)

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

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

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

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

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

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