**NYS000539**
**Submitted: June 9, 2015**

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NL-14-001

January 16, 2014

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

SUBJECT: Proposed License Amendment for Alternate Repair Criteria for Steam Generator Tube Inspection and Repair
Indian Point Unit Number 2
Docket No. 50-247
License No. DPR-26

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc., (Entergy) hereby requests a License Amendment to Operating License DPR-26, Docket No. 50-247 for Indian Point Nuclear Generating Unit No. 2 (IP2). The proposed Technical Specifications (TS) change would revise TS 5.5.7, "Steam Generator (SG) Program" to exclude portions of the SG tube below the top of the SG tubesheet from periodic inspections and plugging by implementing the alternate repair criteria "H*." In addition, TS 5.6.7, "Steam Generator Tube Inspection Report" is also being revised to include additional reporting requirements.

The proposed amendment constitutes a redefinition of the SG tube primary to secondary pressure boundary and defines the safety significant portion of the tube that must be inspected or plugged. Tube flaws detected below the safety significant portion of the tube are not required to be plugged. Allowing flaws in the non-safety significant portion of the tube to remain in service minimizes unnecessary tube plugging and maintains the safety margin of the SGs to perform the safety function to maintain the reactor coolant pressure boundary, maintain reactor coolant flow, and maintain primary to secondary heat transfer.

Entergy has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and has determined that the changes involve no significant hazards considerations, as described in Attachment 1. The marked up pages showing the proposed Technical Specifications changes are provided in Attachment 2.

The proposed TS changes are based on the supporting structural analysis and leakage evaluation completed by Westinghouse Electric Company, LLC (Westinghouse), and included as Enclosure 1 (proprietary) and Enclosure 2 (non-proprietary) as well as additional documentation listed in Attachment 1 to this correspondence (which has been previously submitted to the NRC staff by other utilities.) The information proprietary to Westinghouse in Enclosure 1 is supported by an affidavit signed by Westinghouse, the owner of the information, and included as Enclosure 3. The

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attached affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390. Accordingly, it is requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

A copy of this application and the associated attachments are being submitted to the designated New York State official in accordance with 10 CFR 50.91.

Entergy requests approval of the proposed amendment by February 1, 2015 and an allowance of 30 days for implementation. There are no new commitments being made in this submittal. If you have any questions or require additional information, please contact Mr. Robert Walpole, Manager, Licensing at (914) 254-6710.

I declare under penalty of perjury that the foregoing is true and correct. Executed on January 16, 2014.

Sincerely,

JAV/ai

Attachments:
1. Analysis of Proposed Technical Specifications Changes for Alternate Repair Criteria for Steam Generator Tube Inspection and Repair
2. Marked Up Technical Specifications Pages for Proposed Changes for Alternate Repair Criteria for Steam Generator Tube Inspection and Repair

Enclosures:
1. Westinghouse WCAP-17828-P, Indian Point Unit 2 H* Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (4-Loop Model 44F) (proprietary)
2. Westinghouse WCAP-17828, Indian Point Unit 2 H* Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (4-Loop Model 44F) (non-proprietary)
3. Westinghouse Application for Witholding Proprietary Information from Public Disclosure

cc: Mr. Douglas Pickett, Senior Project Manager, NRC NRR DORL
Mr. William Dean, Regional Administrator, NRC Region 1
NRC Resident Inspector's Office
Mr. Francis J. Murray, Jr., President and CEO, NYSERDA
Ms. Bridget Frymire, New York State Dept. of Public Service

ATTACHMENT 1 TO NL-14-001

# ANALYSIS OF PROPOSED TECHNICAL SPECIFICATIONS CHANGES FOR ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBE INSPECTION AND REPAIR

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

## 1.0 DESCRIPTION

Entergy Nuclear Operations, Inc (Entergy) is requesting an amendment to Operating License DPR-26, Docket No. 50-247 for Indian Point Nuclear Generating Unit No. 2 (IP2). This amendment proposes to revise TS 5.5.7 "Steam Generator (SG) Program" to exclude portions of the SG tube below the top of the SG tubesheet from periodic inspections. This amendment also proposes to include additional reporting requirements in TS 5.6.7, "Steam Generator Tube Inspection Report" which will establish the appropriate reporting criteria for the tubes that require plugging under the proposed alternate repair criteria and a quantification of the operational and accident-induced leakage that could potentially be attributable to the uninspected region of the SG tubes. The proposed change is based on the supporting structural analysis and leakage evaluation completed by Westinghouse Electric Company, LLC (Westinghouse).

## 2.0 PROPOSED CHANGES

TS 5.5.7.c is to be revised to implement the SG tube alternate repair criteria. Wording is being added to specify that the portion of the tube below 18.9 inches (H* value) from the top of the tubesheet is excluded from plugging.

Change TS 5.5.7.c from:

> c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

To:

> c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.
>
> The following SG tube alternate plugging criteria shall be applied as an alternative to the preceding criteria.
>
> Tubes with service-induced flaws located greater than 18.9 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.9 inches below the top of the tubesheet shall be plugged upon detection.

TS 5.5.7.d is being revised to implement the SG tube alternate repair criteria. Wording is being added to specify that the portion of the tube below 18.9 inches (H* value) from the top of the tubesheet is excluded from inspection.

Change TS 5.5.7.d from:

> 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 addition to meeting the requirements of d.1, d.2, and d.3 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. An assessment of degradation 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.

To:

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 18.9 inches below the top of the tubesheet on the hot leg side to 18.9 inches below the top of the tubesheet on the cold leg side, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 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. An assessment of degradation 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.

Additional reporting requirements are being included as TS 5.6.7.i, j and k, which will establish the appropriate reporting criteria for the tubes that require plugging under the proposed alternate repair criteria and a quantification of the operational and accident-induced leakage that could potentially be attributable to the uninspected region of the SG tubes.

Change TS 5.6.7 from:

5.6.7 <u>Steam Generator Tube Inspection Report</u>

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 5.5.7, 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 during the inspection outage for each active degradation mechanism,

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

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

h. The effective plugging percentage for all plugging in each SG.

To:

5.6.7 Steam Generator Tube Inspection Report

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 5.5.7, 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 during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged 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 in each SG,

i. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,

j. The calculated accident leakage rate from the portion of the tubes below 18.9 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 1.75 times the maximum primary to secondary leakage rate, the report should describe how it was determined, and

k. The results of monitoring for tube displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

## 3.0 BACKGROUND

IP2 is a four loop Westinghouse designed plant with Model 44F replacement steam generators that were installed in 2000. Each SG has 3214 thermally treated Alloy 600 tubes; each tube is fully expanded through the tubesheet thickness using a hydraulic expansion process. The six tube support plates are fabricated from 405 stainless steel and include broached quatrefoil tube holes in a square pitch array. A total of 34 tubes are currently plugged throughout the four steam generators.

The SG tubes in pressurized water reactors have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they act as a heat transfer surface between the primary and secondary systems to remove heat from the primary system. In addition, the SG tubes isolate the radioactive fission products in the primary coolant from the secondary system.

Steam generator tube integrity is necessary in order to satisfy the tubing's safety functions. Maintaining tube integrity ensures that the tubes are capable of performing their intended safety functions consistent with the plant licensing basis, including applicable regulatory requirements. Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as wear. These degradation mechanisms can impair tube integrity if they are not managed effectively.

The industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP) developed a series of EPRI Guidelines. These guidelines include

- "Pressurized Water Reactor Steam Generator Examination Guidelines" (Reference 1),
- "Steam Generator Integrity Assessment Guidelines" (Reference 2)

These EPRI Guidelines, along with NEI 97-06 (Reference 3), tie the entire Steam Generator Program together, while defining a comprehensive, performance based approach to managing SG performance. As a result of interactions between the industry and the NRC, the Technical Specifications Task Force (TSTF) prepared TSTF-449, Revision 4, "Steam Generator Tube Integrity" (Reference 4). The NRC staff approved TSTF-449, Revision 4 and it was posted for adoption by licensees in the NRC Federal Register Notice of Availability published on May 6, 2005 (70 FR 24126). This approach was adopted by IP2 after issuance of Amendment 251 (Reference 5) on February 13, 2007.

Further industry initiatives resulted in development of the technical justifications reported in WCAP-17091-P, Revision 2, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F/51F)" (Reference 6) and WCAP-17345-P, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3 Loop Model 44F/Model 51F) (Reference 7). Portions of information in these WCAPs directly apply to IP2

H* development, as do the documents in Table 1. These documents have been previously docketed in other H* submittals, and are hereby being incorporated by reference into this amendment request submittal. An IP2 specific structural analysis and leakage evaluation based on contact pressure between the tube and tubesheet was performed in WCAP-17828-P, "Indian Point Unit 2 H* Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (Model 44F-4-Loop) (Reference 8), and included as Enclosure 1 (proprietary) and Enclosure 2 (non-proprietary).

The alternate repair criteria requires that tubes with service induced flaws located in the portion of the tube from the top of the tubesheet to a distance of H* below the top of the tubesheet be plugged upon detection. Tubes with service-induced flaws located greater than a distance of H* below the top of the tubesheet are not required to be plugged. When applicable, the alternate repair criteria exclude that portion of each SG tube from a distance of H* below the top of the tubesheet to the tube-to-tubesheet weld at the bottom of the tubesheet from inspections required by specification 5.5.7.d of TS 5.5.7 "Steam Generator (SG) Program."

The analyses in the referenced documents demonstrate that the integrity of steam generator tubes as part of the RCPB is not dependent upon the portion of the tube near the tube end welds at the bottom of the tubesheet. Application of the alternate repair criteria reduces the number of SG tubes that are required to be plugged while maintaining established margins of safety for normal and accident conditions.

Catawba Nuclear Station, Unit 2, (Catawba) reported indications of cracking following nondestructive eddy current examination of the steam generator tubes during their fall 2004 outage. NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," (Reference 9), provided industry notification of the Catawba issue. IN 2005-09 noted that Catawba reported crack-like indications in the tubes approximately seven inches below the top of the hot leg tubesheet in one tube, and just above the tube-to-tubesheet welds in a region of the tube known as the tack expansion in several other tubes. Indications were also reported in the tube-end welds, also known as tube-to-tubesheet welds, which join the tube to the tubesheet.

The experience at Catawba (as noted in IN 2005-09) shows the importance of monitoring all tube locations (such as bulges, dents, dings, and other anomalies from the manufacture of the SGs) with techniques capable of finding potential forms of degradation that may be occurring at these locations such as those discussed in NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections" (Reference 10). Since the IP2 Westinghouse Model 44F SGs were fabricated with Alloy 600 thermally treated tubes, similar to the Catawba Unit 2 Westinghouse Model D5 SGs, a potential exists for IP2 to identify tube indications similar to those reported at Catawba within the hot leg tubesheet region.

Potential inspection plans for the tube and tube welds underwent extensive industry discussion in March of 2005. The findings in the Catawba Unit 2 SG tubes present three distinct issues with regard to the SG tubes at Indian Point Unit 2:

1) Indications in internal bulges and over-expansions within the hot leg tubesheet,
2) Indications at the elevation of the tack expansion, and,
3) Indications in the tube-to-tubesheet welds and propagation of these indications into adjacent tube material.

Prior to each SG tube inspection, a degradation assessment, which includes a review of operating experience, is performed to identify degradation mechanisms that have a potential to be present in the IP2 SGs. The degradation assessment also verifies that the eddy current techniques utilized are capable of detecting those flaw types that are identified in the degradation assessment. Based on operating experience, IP2 revised the SG inspection plans for the refueling outage 17 (RO17) inspections in the spring of 2006 to include a 20% sample of overexpansions (OXP), bulges (BLG) and dents (DNT) within the tubesheet region on the hot leg side. No degradation was detected as a result of these inspections.

The inspections in RO17 and RO19 were based on the guidance contained in the EPRI PWR Steam Generator Examination Guidelines, Revisions 6 & 7, and IP2 Technical Specification 5.5.7 "Steam Generator (SG) Program." According to the EPRI SG Examination Guidelines, the inspection plan is expanded if necessary due to confirmed degradation in the region required to be examined (i.e. a tube crack). At IP2, anti-vibration bar (AVB) wear is the only existing degradation mechanism, and no crack-like indications have ever been detected at IP2.

Based on the IP2 inspections, no service induced flaws are anticipated in the near future. However, in the event of any potential issues, and so as not to unnecessarily plug steam generator tubes, Entergy is proposing changes to TS 5.5.7 and TS 5.6.7 to adopt the H* methodology, so as to limit SG tube inspection and plugging to the safety significant portion of the tubes.

## 4.0 Summary of Licensing Basis (H* Analysis)

In addition to WCAP-17828-P, the technical justification of H* for the Indian Point Unit 2 SGs is comprised of a number of reports that have been previously provided to the Nuclear Regulatory Commission (NRC) staff to support its licensing actions for domestic utilities (See Table 1). A synopsis of each of these documents is provided below:

Item 1: This is a principal licensing document. It responds to the last NRC staff request for additional information (RAI) and is key to licensing the permanent application of H*. The document is written for the Surry Units 1 and 2 Model 51F steam generators because they were the first permanent H* license request for the 3-loop plants in the H* fleet. Responses to the RAIs for the Indian Point Unit 2 Model 44F SGs are characteristically the same.

Item 2: This is a principal licensing document. In response to the NRC staff RAI relating to Item 3 (see also Items 4 and 5), Westinghouse applied a structural model, the $C^2$ model, entirely different than that discussed in Item 3, to calculate the tube-to-tubesheet contact pressures. The $C^2$ model avoids the issue of tubesheet bore eccentricity and how to account for it in the analysis (see Items 4, 4a and 5). This report describes in detail the application of the $C^2$ model to calculate the final H* depth and is the final basis of the recommended H* depth.

Item 3: This is a principal licensing document. WCAP-17091-P describes the original approaches and models for both structural and leakage analyses that are the technical calculation basis to determine the H* distance and the applicable leakage factor. The original structural analysis utilized the classical thick-shell equations to calculate tube-to-tubesheet contact pressure. The approach to leakage analysis is essentially the same as that in the thick shell model. The report contains an evaluation of the statistical variability in the coefficient of thermal expansion properties of the SA-508 tubesheet material and of Alloy 600, the tube material. While Item 2 provides the most current technical justification for H*, this report is a principal reference that supports Item 2.

Item 4 and 4a: These documents were considered by the NRC staff to be an important licensing document; they contain responses to NRC staff questions related to a technical issue that prevented permanent licensing of H* at the end of calendar year (CY) 2009. The technical issue concerned the treatment of tubesheet bore eccentricity in the application of the thick-shell equations to calculate tube-to-tubesheet contact pressures. This issue was finally addressed in Item 5. However the issue of eccentricity of the tubesheet bores is unrelated to the current technical basis for H*, which relies on application of a local structural model, the $C^2$ model described in detail in Item 2. The $C^2$ model does not require separate accounting for tubesheet bore eccentricity because application of the boundary conditions to the $C^2$ model automatically accounts for tubesheet bore eccentricity. Item 4a was written to address Model F and D5 SGs. The responses to the RAI in Item 4a apply to the Model 44F and Model 51F SGs.

Item 5: This document was considered by the NRC staff to be an important licensing document because it finalizes the technical discussion in Item 4. However, the information contained in the document has no impact on the final technical justification of H*. The final H* analysis utilizes a local structural model, the $C^2$ model, that replaced the original approach using thick-shell formulation (see Item 3) to calculate contact pressures between the tube and the tubesheet. This document concludes that accounting for tubesheet bore eccentricity in the thick-shell approach has a very small, but conservative, effect on the calculated contact pressures.

Item 6: This is a principal licensing document. It describes the approach to probabilistic analysis to determine the high confidence values of H* such as the 0.95 probability at 95% confidence value and also the analysis of leakage factors for the limiting accident. It is noted that the industry standard for probabilistic analysis is 0.95 probability at 50% confidence; however, the NRC staff imposed the new requirement of 0.95 probability at 95% confidence for H*. In regard to leakage analysis, the NRC staff questioned the definition of the limiting accident with respect to leakage. The result of this question was to also consider the feedwater line break (FLB) accident as a heat-up event in determining the leakage factor. The FLB is not part of the licensing basis for Indian Point Unit 2.

Item 7: This is a principal licensing document. The NRC staff required that utilities wishing to implement H* monitor the tubes for slippage. The NRC continues to require a commitment to monitor for tube slippage. This document provides the guidelines for monitoring for tube slippage.

Item 8: This is a principal licensing document. The key parameters in the H* technical justification are the coefficients of thermal expansion (CTE) of the tube and tubesheet materials. In 2007, the NRC staff provided a data point for tubesheet CTE that suggested that the American Society of Mechanical Engineers Boiler and Pressure Vessel Code values for CTE were incorrect. This led to a test program to determine the CTEs of the tube and tubesheet material and to establish the statistical variability of these CTEs. The document provides the statistical analysis of the test data which is used as a prime input to the probabilistic analysis of H*. Note that the same document is provided as Appendix B of Item 3.

Item 9: This is a principal licensing document. This document describes the analysis of crevice pressure distribution test data and the impact of the crevice pressure tests on H*. Early H* technical justifications assumed that the crevice between the tube and the tubesheet is at the secondary side saturation pressure. The tests showed that this assumption was not supported and that the crevice pressure has an axial distribution between the elevation of the flaw and the top of the tubesheet. This test data significantly changed the approach to calculating the H* distance. The crevice pressure distribution is reflected in the current technical justification of H*.

Item 10: This document has historical significance. The justification of H* discussed in Item 3 showed that positive contact pressure between the tube and tubesheet existed at the top of the tubesheet. Consequently, the location of the bottom of the expansion transition was a potentially important factor in the final H* depth. This document evaluates the acceptable depth of the expansion transition relative to the H* calculations described in Item 3. It is also necessary to show that there are no locations of the expansion transition that are more than 1 inch from the top of the tubesheet.

Item 11: This document is primarily background information and is not considered to be a principal licensing document. The analysis compares the 0.95 probability at 95% confidence H* depths based on the analysis results from Item 3 and introduces the Monte Carlo sampling technique using the uncertainty surface developed in Item 3 (Section 8). Historically, the use of this paper was to justify interim application of H* for one cycle. The longer term impact of this paper is that the same Monte Carlo process was applied for the final H* analysis documented in Item 2.

Item 12: These documents correct minor errors in Item 6 and are provided here only for completeness.

**Table 1**
**Summary of Documents that Support the Technical Justification of H***

| Item | Document Number | Document Title | Date | Notes |
|---|---|---|---|---|
| 1 | LTR-SGMMP-11-29 P-Attachment, Rev. 1 | Response to USNRC Request for Additional Information Regarding the Surry Units 1&2 License Amendment Request For Permanent Application of the Alternate Repair Criterion H* | 1/24/2012 | 1 |
| 2 | WCAP-17345-P, Revision 2 | H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3 Loop Model 44F/Model 51F) | June 2011 | 1 |
| 3 | WCAP-17091-P, Revision 2 | H*; Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F/51F) | June 2009 | 1,5 |
| 4 | LTR-SGMP-10-33 P-Attachment | H*: Response to NRC Questions Regarding Tubesheet Bore Eccentricity | Sept. 2010 | 1,2 |
| 4a | LTR-SGMP-09-109 P-Attachment | NRC Request for Additional Information in H*; RAI #4; Model F and Model D5 Steam Generators | August 2009 | 1, 2 |
| 5 | LTR-SGMP-10-78 P-Attachment | Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and their Relative Importance to H* | Sept. 2010 | 1,3 |
| 6 | LTR-SGMP-09-108 P-Attachment, Revision 0 | Response to NRC Request for Additional Information on H*; Model 44F and Model 51F5 Steam Generators | 8/28/2009 | 1,5 |
| 7 | LTR-SGMP-09-140 | H*: Guideline for Monitoring for Tube Slippage Using the Bobbin Probe Data | 9/22/2009 | 1, 7 |
| 8 | WEST-13-402, Revision 1 | An Evaluation of the Statistical Variability in Coefficient of Thermal Expansion Properties of SA-508 and Alloy-600 | 12/24/2008 | 1, 8 |
| 9 | SG-CDME-07-2 P-Attachment | White Paper Addressing Changes to B* and H* Analysis due to New Crevice Pressure and Divider Plate Information | 2/2/2007 | 1, 8 |
| 10 | LTR-SGMP-09-111 P-Attachment Revision 1 | Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H* | Sept. 2010 | 4 |
| 11 | LTR-SGMP-09-104-P Attachment, Revision 1 | White Paper on Probabilistic Assessment of H* | 8/13/2009 | 6 |
| 12 | LTR-SGMP-09-108 Errata, LTR-SGMP-09-137 | Miscellaneous Errata. | September 2009 | 5 |

Notes:
1. Principal Licensing Basis document.
2. Final Response to NRC RAI on Eccentricity (Single Technical Issue).
3. Partial Response to NRC RAI on Eccentricity (Single Technical Issue).
4. Provides background for technical development or was requested by the USNRC.
5. 2009 Technical Justification; required to complete current technical basis.
6. Provides 95/95 probability H* values requested by NRC; industry requirement is 95/50.
7. Response to commitment required by NRC.
8. Essential technical information to support H* structural and/or leakage analysis.

## 5.0 Technical Evaluation

The proposed Amendment includes changes for alternate repair criteria to adopt H* methodology for SG tube inspection and plugging.

To preclude unnecessarily plugging tubes in the IP2 steam generators, an evaluation was performed to identify the safety significant portion of the tube within the tubesheet necessary to maintain structural and leakage integrity in both normal and accident conditions. Tube inspections will be limited to identifying and plugging degradation in the safety significant portion of the tubes. The technical evaluation for the inspection and plugging methodology for IP2 is provided in Reference 8, and supplemented by the documents listed previously in Table 1. This evaluation is based on the use of finite element model structural analysis and a bounding leak rate evaluation based on contact pressure between the tube and the tubesheet during normal and postulated accident conditions. The limited tubesheet inspection criteria were developed for the tubesheet region of the IP2 Model 44F steam generators considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The limited tubesheet inspection criteria were selected to prevent tube burst and axial separation due to axial pullout forces acting on the tube and to ensure that the accident induced leakage limits are not exceeded. The H* analysis provides technical justification for limiting the inspection in the tubesheet expansion region to less than the full depth of the tubesheet.

The determination of the safety significant portion of the tube within the tubesheet is based upon evaluation and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in the H* analysis. The tube-to tubesheet radial contact pressure provides resistance to tube pullout and resistance to leakage during plant operation and transients.

Reference 8 redefines the primary pressure boundary. The tube-to-tubesheet weld no longer functions as a portion of this boundary. The hydraulically expanded portion of the tube into the tubesheet over the H* distance now functions as the primary pressure boundary in the area of the tube and tubesheet, maintaining the structural and leakage integrity consistent with NEI 97-06 (Reference 3) SG tube integrity performance criteria over the full range of steam generator operating conditions, including the most limiting accident conditions. The supporting Westinghouse analysis determined that degradation in tubing below this safety significant portion of the tube does not require inspection or plugging. The inspection of the safety significant portion of the tubes provides a high level of confidence that the structural and leakage performance criteria are maintained during normal operating and accident conditions.

The constraint that is provided by the tubesheet precludes tube burst from cracks within the tubesheet. The criteria for tube burst described in NEI 97-06 (Reference 3) and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," (Reference 11) are satisfied due to the constraint provided by the tubesheet and the tube-to-tubesheet joint. Through application of the limited tubesheet inspection scope as described below, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur.

Primary-to-secondary leakage from tube degradation in the tubesheet area is assumed to occur in several design basis accidents: Main Steam Line Break (MSLB), Locked Rotor, and Control Rod Ejection. The radiological dose consequences associated with this assumed leakage are evaluated to ensure that they remain within regulatory limits (e.g., 10 CFR 50.67). The accident

induced leakage performance criteria are intended to ensure the primary-to-secondary leak rate during any accident does not exceed the primary-to-secondary leak rate assumed in the accident analysis.

As discussed in References 6 and 8, the leak rate ratio (accident induced leak rate to operational leak rate) is a product of the pressure differential subfactor and the viscosity subfactor using the Darcy flow equation. For the postulated MSLB event, a plant cool down event would occur and the subsequent temperatures in the reactor coolant system (RCS) would not be expected to exceed the temperatures at plant no-load conditions. An increase in leakage would not be expected to occur as a result of the temperature change and the viscosity subfactor can be conservatively set equal to 1.0. Therefore, the increase in leakage would only be a function of the increase in primary-to-secondary pressure differential. The resulting leak rate ratio for the MSLB is 1.75.

The current licensing basis MSLB accident analysis calculations assume a primary to secondary leakage equivalent to the TS leak rate limit of 150 gallons per day (gpd) (0.11 gpm) through any one steam generator. However, the leak rate is administratively limited to 75 gpd/SG or about 0.055 gpm. When this leakage is multiplied by the MSLB leak rate factor of 1.75, the maximum primary to secondary accident induced leak rate is about 132 gpd (0.096 gpm) and is bounded by the value of 150 gpd (0.11 gpm) through the faulted SG used in the MSLB accident analysis.

The other design basis accidents, such as the postulated locked rotor event and the control rod ejection event, are conservatively modeled using design specification transients which result in increased temperatures in the steam generator hot and cold legs for a period of time. However, the length of time that a plant with Model 44F steam generators will exceed the normal operating differential pressure across the tubesheet is less than 30 seconds for the locked rotor event, and less than 10 seconds for the control rod ejection event. As the accident induced leakage performance criteria is defined in gallons per minute, the leak rate for a locked rotor and a control rod ejection event can be integrated over a minute to compare to the limit. Time integration permits an increase in acceptable leakage during the time of peak pressure differential by approximately a factor of two for the locked rotor event because of the short duration (less than 30 seconds) of the elevated pressure differential. This translates into an effective reduction in leakage factor of two for the locked rotor event. Therefore, for the locked rotor event, the leakage factor of 1.67 for IP2 is adjusted downward to a factor of 0.84. Similarly, for the control rod ejection event, the duration of the elevated pressure differential is less than 10 seconds. Thus, the peak leakage factor may be reduced by a factor of six from 2.36 to 0.39. Due to the short duration of the transients above normal operating pressure differential, no leakage factor is required for the locked rotor and control rod ejection events (i.e., the leakage factor is under 1.0 for both transients). Thus, MSLB is the limiting accident and 1.75 remains the limiting leak rate factor for IP2.

WCAP-17091-P (Reference 6), section 9.8, provides a review of leak rate susceptibility due to tube slippage and concluded that the tubes are fully restrained against motion under very conservative design and analysis assumptions such that tube slippage is not a credible event for any tube in the bundle. The proposed TS changes will require monitoring for steam generator tube slippage be part of the IP2 steam generator tube inspection program and the results of slippage monitoring be included in the reporting requirements of TS 5.6.7, "Steam Generator Tube Inspection Report."

## 6.0 REGULATORY ANALYSIS

### 6.1 No Significant Hazards Consideration

This amendment request proposes to revise TS 5.5.7, "Steam Generator (SG) Program" to exclude portions of the SG tube below the top of the SG tubesheet from periodic inspections and plugging by implementing the alternate repair criteria "H*." This amendment also proposes to include additional reporting requirements in TS 5.6.7, "Steam Generator Tube Inspection Report" which will establish the appropriate reporting criteria for the tubes that require plugging under the proposed alternate repair criteria and a quantification of the operational and accident-induced leakage that could potentially be attributable to the uninspected region of the SG tubes.

Entergy Nuclear Operations, Inc. (Entergy) has evaluated the safety significance of the proposed change to the Indian Point 2 Technical Specifications (TS) according to the criteria of 10 CFR 50.92, "Issuance of Amendment". Entergy has determined that the subject changes do not involve a Significant Hazards Consideration, as discussed below:

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

Response: No.

The proposed change excludes the lower portion of steam generator tubes from inspection by implementing the alternate repair criteria H* and does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change has no significant effect upon accident probabilities or consequences.

Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed change to the steam generator tube inspection and repair criteria are the steam generator tube rupture (SGTR), the main steam line break (MSLB), Locked Rotor and Control Rod Ejection.

At normal operating pressures, leakage from Primary Water Stress Corrosion Cracking (PWSCC) below the proposed limited inspection depth 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.

For the SGTR event, the required structural integrity margins of the steam generator tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the tube-to-tubesheet joint. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side. The structural margins against burst, as discussed in Regulatory Guide (RG) 1.121,"Bases for Plugging Degraded PWR Steam Generator Tubes," (Reference 11) and NEI 97-06, "Steam Generator Program Guidelines"' (Reference 3) are maintained for both normal and postulated accident conditions. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

The probability of a Steam Line Break, Locked Rotor, and Control Rod Ejection are not affected by the potential failure of a SG tube, as the failure of a tube is not an initiator for any of these events.

In the supporting Westinghouse analyses, leakage is modeled as flow through a porous medium via the use of the Darcy equation. The leakage model is used to develop a relationship between allowable leakage and leakage at accident conditions that is based on differential pressure across the tubesheet and the viscosity of the fluid. A leak rate ratio was developed to relate the leakage at operating conditions to leakage at accident conditions. The fluid viscosity is based on fluid temperature and it has been shown that for the most limiting accident, the fluid temperature does not exceed the normal operating temperature. Therefore, the viscosity ratio is assumed to be 1.0 and the leak rate ratio is a function of the ratio of the accident differential pressure and the normal operating differential pressure.

The leakage factor of 1.75 for IP2 for a-postulated MSLB, has been calculated as shown in the supporting Westinghouse analysis. IP2 will apply a factor of 1.75 to the normal operating leakage associated with the tubesheet expansion region in the Condition Monitoring Assessment and Operational Assessment. Through application of the limited tubesheet inspection scope, the administrative leakage limit of 75 gpd provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. No leakage factor will be applied to the Locked Rotor or Control Rod Ejection due to their short duration, since the calculated leak rate ratio is less than 1.0. Therefore, the proposed change does not result in a significant increase in the consequences of these accidents.

For the Condition Monitoring Assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 1.75 and added to the total leakage from any other source and compared to the allowable MSLB leakage limit. For the Operational Assessment, the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 1.75 and compared to the observed operational leakage. As noted above, an administrative limit of 75 gpd has been established at IP2 to assure that the allowable accident induced leakage is not exceeded.

Based on the above, the performance criteria of NEI 97-06 and Regulatory Guide (RG) 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change excludes the lower portion of steam generator tubes from inspection by implementing the alternate repair criteria (H*). The proposed change does not introduce any new equipment, create new failure modes for existing equipment, or create any new limiting single failures resulting from tube degradation. The proposed change does not affect the design of the SGs or their method of operation. In addition, the proposed change does not impact any other plant system or component. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in accident analyses. 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 amendment involve a significant reduction in a margin of safety?**

Response: No.

The proposed change defines the safety significant portion of the SG tubing that must be inspected and repaired. WCAP-17828-P identifies the inspection depth below which any type of degradation is shown to have no impact on the steam generator tube integrity performance criteria in NEI 97-06. The proposed change does not affect tube design or operating environment. The proposed change will continue to require monitoring of the physical condition of the SG tubes but will limit inspection within the tubesheet to the portion of the tube from the top of the tubesheet to a distance H* below the top of the tubesheet.

The proposed change maintains the required structural margins of the SG tubes for both normal and accident conditions. For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, the supporting Westinghouse analyses define a length 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 limited hot and cold leg tubesheet inspection criteria will preclude unacceptable primary to secondary leakage during all plant conditions. The MSLB leak rate factor for IP2 is 1.75. Multiplying the IP2 administrative leak rate limit of 75 gpd/SG by this factor shows that the primary-to-secondary leak rate during a postulated SLB is not exceeded. Therefore, the proposed change does not involve a significant reduction in any margin of safety.

Based on the above, Entergy concludes that the proposed amendment to the Indian Point 2 Technical Specifications presents 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.

6.2 Applicable Regulatory Requirements / Criteria

The General Design Criteria (GDC) which formed the bases for Indian Point 2 design was published by the Atomic Energy Commission (AEC) in the Federal Register of July 11, 1967 and subsequently made part of 10 CFR 50. The application of the AEC proposed GDC to IP2 is contained in the UFSARs. Appendix A of 10 CFR Part 50 GDC differ both in numbering and content from the AEC for IP2. The following provides discussion of the proposed change on the capability of IP2 for continued compliance with the 1967 GDCs.

The regulatory requirements applicable to SG tube integrity are the following:

*1967 GDC-9 Reactor Coolant Pressure Boundary* - The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.

The proposed change does not alter the SG design, fabrication, erection or testing.

*1967 GDC-16 Monitoring Reactor Coolant Leakage* - Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.

The proposed change does not alter the means of detecting leakage from the reactor coolant pressure boundary.

*1967 GDC-33 Reactor Coolant Pressure Boundary Capability* -The RCPB shall be capable of accommodating without rupture the static and dynamic load imposed on any boundary component

as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

The proposed change excludes from inspection those portions of the steam generator tubes that are not safety significant with respect to maintaining the reactor coolant pressure boundary. Structural analyses demonstrate that the safety significant portion of the steam generator tube within the tubesheet maintains the capability to accommodate, without rupture, the sudden release of energy into the coolant.

*1967 GDC-34 RCPB Rapid Propagation Failure Prevention* - The RCPB shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure.

The proposed change does not alter the SG design or operation.

*1967 GDC-36 RCPB Surveillance* - RCPB components shall have provisions for inspection, testing, and surveillance of criteria areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime.

The proposed change does not alter provisions for inspection, testing, or surveillance of criteria applicable to the safety significant portions of steam generator tubes credited as part of the reactor coolant pressure boundary.

*10 CFR 50.55a, Codes and Standards - Section (b), ASME Code* - c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

The proposed change and the Steam Generator Program requirements which underlie it are in full compliance with the ASME Code. The proposed technical specifications are based on conservative analysis which ensure tube integrity and, therefore, compliance with the ASME Code.

*10 CFR 50.65 Maintenance Rule* - Each holder of a license to operate a nuclear power plant under 50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in 50.82(a)(1), this section only shall apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended functions.

Under 10 CFR 50.65, the Maintenance Rule, licensees classify SGs as risk significant components because they are relied upon to remain functional during and after design basis events. SGs are to be monitored under 10 CFR 50.65(a)(2) against industry established performance criteria. Meeting

the performance criteria of NEI 97-06, Revision 3 (Reference 3), provides reasonable assurance that the SG tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary. These are:

- The structural integrity performance criterion: *All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown, and all anticipated transients included in the design specification) and design basis accidents.*

The proposed change defines the portion of the tube that is engaged in the tubesheet from the secondary face that is required to maintain structural and leakage integrity over the full range of SG operating conditions, including the most limiting accident conditions. The H* distance for the tubesheet region has been developed to meet the above criteria. The structural criterion regarding tube burst is inherently satisfied because the constraint provided by the tubesheet to the tube prohibits burst. WCAP-17828-P, Rev. 0, (Reference 8) determined the 95/95 whole bundle depth of 18.9 inches from the top of the tubesheet that will preclude tube pullout for all plant conditions. As such, the IP2 inspection program provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

- The 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.*

The reactor coolant system primary-to-secondary leakage through any one steam generator shall be administratively limited to 75 gpd.

### 6.3 Environmental Considerations

IP2 has evaluated the proposed amendment for environmental considerations. The 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, and 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 for in 10 CFR 51.22(c) (9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

## 7.0 PRECEDENCE

This request is similar to the following license amendment requests approved by the NRC staff as noted:

Virginia Electric and Power Company (Dominion) Surry Power Station Units I and 2 License Amendment Request Permanent Alternate Repair Criteria For Steam Generator Tube Inspection and Repair, dated July 28, 2011, (Reference 12) as approved by NRC letter, Surry Power Station Unit Nos. 1 and 2, Issuance of Amendments Regarding Virginia

Electric and Power Company License Amendment Request for Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair (TAC Nos. ME6803 and ME6804), dated April 17, 2012 (Reference 13)

Catawba Nuclear Station, Units 1 and 2 - Proposed Technical Specifications (TS) Amendment - TS 3.4.13, "RCS Operational LEAKAGE," TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.8, "Steam Generator Tube Inspection Report" - License Amendment Request to Revise TS for Permanent Alternate Repair Criteria, dated June 30, 2011, (Reference 14) as approved by NRC letter, Catawba Nuclear Station, Units 1 and 2, Issuance of Amendments Regarding Technical Specification Amendments for Permanent Alternate Repair Criteria for Steam Generator Tubes (TAC Nos. ME6670 and ME6671), dated March 12, 2012 (Reference 15)

Turkey Point Nuclear Generating Station Unit Nos. 3 and 4- Permanent Alternate Repair Criteria for Steam Generator Expansion Region, dated April 30, 2012 (Reference 16) as approved by NRC letter, Turkey Point Nuclear Generating Station Unit Nos. 3 and 4- Issuance of Amendments regarding Permanent Alternate Repair Criteria for Steam Generator Tubes (TAC NOS. ME8515 and ME8516)", dated November 5, 2012 (Reference 17)

## 8.0 REFERENCES

1. EPRI 1013706 "Pressurized Water Reactor Steam Generator Examination Guidelines" Rev. 7, October 2007
2. EPRI 1019038 "Steam Generator Integrity Assessment Guidelines" November 2009
3. NEI 97-06 "Steam Generator - Program Guidelines" Revision 3, January 2011 (ADAMS Accession No. ML 111310708)
4. Technical Specification Task Force (TSTF) Letter to NRC, TSTF-05-05 "TSTF-449, Revision 4, "Steam Generator Tube Integrity" April 14, 2005 (ADAMS Accession No. ML051090200)
5. NRC letter, "Indian Point Nuclear Generating Unit Nos. 2 and 3 - Issuance of Amendments Re: Steam Generator Tube Integrity Technical Specification Based on Technical Specification Task Force (TSTF) Document TSTF-449, "Steam Generator Tube Integrity" (TAC Nos. MD0083, MD0084, MD2178, and MD2179)," February 13, 2007
6. Westinghouse Electric Company WCAP-17091-P, Revision 2, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F/51F)" June, 2009
7. Westinghouse Electric Company WCAP-17345-P, Revision 2, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3-Loop Model 44F/Model 51 F)" June 2011
8. Westinghouse Electric Company WCAP-17828, Indian Point Unit 2 H* Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (Model 44F-4-LOOP)
9. NRC Information Notice 2005-09 "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds" April 7, 2005
10. NRC Generic Letter 2004-01 "Requirements for Steam Generator Tube Inspections," August 30, 2004
11. NRC Regulatory Guide 1.121 "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976
12. Virginia Electric and Power Company (Dominion) letter to NRC, "Surry Power Station Units 1 and 2 License Amendment Request Permanent Alternate Repair Criteria for Steam

Generator Tube Inspection and Repair" July 28, 2011 (ADAMS Accession No. ML11215A058)

13. NRC letter to Surry Power Station, "Surry Power Station, Unit Nos. 1 and 2, Issuance of Amendments Regarding Virginia Electric and Power Company License Amendment Request for Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair (TAC Nos. ME6803 and ME6804)" April 17, 2012 (ADAMS Accession No. ML12109A270)
14. Catawba Nuclear Station letter to NRC, "Proposed Technical Specifications (TS) Amendment - TS 3.4.13, "RCS Operational LEAKAGE," TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.8, "Steam Generator Tube Inspection Report" License Amendment Request to Revise TS for Permanent Alternate Repair Criteria, June 30, 2011 (ADAMS Accession No. MLI 11188A107)
15. NRC Letter to Catawba Nuclear Station, "Catawba Nuclear Station, Units I and 2, Issuance of Amendments Regarding Technical Specification Amendments for Permanent Alternate Repair Criteria for Steam Generator Tubes (TAC Nos. ME6670 and ME6671)" March 12, 2012 (ADAMS Accession No. ML 12054A692)
16. Florida Power and Light Letter to NRC, "Permanent Alternate Repair Criteria (H*) for Steam Generator Expansion Region," dated April 30, 2012
17. NRC Letter to Turkey Point Nuclear Station, "Turkey Point Nuclear Generating Station Unit Nos. 3 and 4-Issuance of Amendments regarding Permanent Alternate Repair Criteria for Steam Generator Tubes (TAC NOS. ME8515 and ME8516)", November 5, 2012 (ADAMS Accession No. ML 12292A342)

ATTACHMENT 2 TO NL-14-001

# MARKED UP TECHNICAL SPECIFICATIONS PAGES FOR PROPOSED CHANGES FOR ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBE INSPECTION AND REPAIR

Changes indicated by ***Bold/Italics*** for additions and ~~strikeout~~ for deletions

Unit 2 Affected Pages:

5.5-7
5.6-5

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

## 5.5 Programs and Manuals

5.5.6 Inservice Testing Program (continued)

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities,

c. The provisions of SR 3.0.3 are applicable to inservice testing activities, and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

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 of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged, to confirm that the performance criteria are being met.

b. 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 retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and 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

## 5.5 Programs and Manuals

### 5.5.7 Steam Generator (SG) Program (continued)

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.

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. Leakage is not to exceed 150 gpd 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. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

***The following SG tube alternate plugging criteria shall be applied as an alternative to the preceding criteria.***

***Tubes with service-induced flaws located greater than 18.9 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 18.9 inches below the top of the tubesheet shall be plugged upon detection.***

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 ***18.9 inches below the top of the tubesheet on the hot leg side to 18.9 inches below the top of the tubesheet on the cold leg side,***~~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 addition to meeting the requirements of d.1, d.2, and d.3 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. An assessment of degradation shall be performed to determine the type and

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For information only



5.5.7 Steam Generator (SG) Program (continued)

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 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, 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 48 effective full power months or two 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 refueling outage (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-line indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operations primary to secondary LEAKAGE

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report", April 1995;

9. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985;

10. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model
Using the NOTRUMP Code", August 1985; and

11. WCAP-10054-P-A, Addendum 2, Revision 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection Into the Broken Loop and Cosi Condensation Model", July 1997.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency
Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided
to the NRC upon issuance for each reload cycle.

5.6.6 Post Accident Monitoring Report

When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following
14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Steam Generator Tube Inspection Report

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 5.5.7, 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,

5.6 Reporting Requirements

5.6.7 Steam Generator Tube Inspection Report (continued)

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

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

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

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

h. The effective plugging percentage for all plugging in each SG~~.~~,

***i. The primary to secondary leakage rate observed in each SG (if it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report,***

***j. The calculated accident leakage rate from the portion of the tubes below 18.9 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident leakage rate from the most limiting accident is less than 1.75 times the maximum primary to secondary leakage rate, the report should describe how it was determined, and***

***k. The results of monitoring for tube displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.***

ENCLOSURE 3 TO NL-14-001

WESTINGHOUSE APPLICATION FOR WITHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Westinghouse

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Proj letter: IPP-14-1

CAW-14-3880

January 9, 2014

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-17828-P, Revision 0, "Indian Point Unit 2 Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (Model 44F – 4-Loop)" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-14-3880 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, 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 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Entergy Nuclear Northeast.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse Affidavit should reference CAW-14-3880 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, Suite 310, 1000 Westinghouse Drive, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

Enclosures

CAW-14-3880

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared James A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

James A. Gresham, Manager
Regulatory Compliance

Sworn to and subscribed before me
this 9th day of January 2014

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Anne M. Stegman, Notary Public
Unity Twp., Westmoreland County
My Commission Expires Aug. 7, 2016
MEMBER, PENNSYLVANIA ASSOCIATION OF NOTARIES

(1) I am Manager, Regulatory Compliance, in Engineering, Equipment and Major Projects, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component

may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-17828-P, Revision 0, "Indian Point Unit 2 Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (Model 44F – 4-Loop)" (Proprietary), dated January 2014, for submittal to the Commission, being transmitted by Entergy Nuclear Northeast letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for Indian Point Unit 2 is that associated with Westinghouse's request for NRC approval of WCAP-17828-P, and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Obtain NRC approval of WCAP-17828-P, "Indian Point Unit 2 Alternate Repair Criteria for the Tubesheet Hydraulic Expansion Region (Model 44F – 4-Loop)."

(ii) License the H* Steam Generator Tube Alternate Repair Criteria.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of licensing the H* Steam Generator Tube Alternate Repair Criteria.

(ii) Westinghouse can sell support and defense of the H* criteria.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

## PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

## COPYRIGHT NOTICE