



APR 30 2012

L-2012-077
10 CFR 50.90

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Re: Turkey Point Nuclear Generating Station Units 3 and 4
Docket Nos.50-250 and 50-251
License Amendment Request No. 220
Permanent Alternate Repair Criteria (H*) for Steam Generator Expansion Region

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests an amendment to the Technical Specifications (TS) of Renewed Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4.

This amendment request proposes to permanently revise TS 6.8.4.j, Steam Generator (SG) Surveillance Program, to exclude portions of the SG tube below the top of the SG tubesheet from periodic tube inspections. Application of the supporting structural analysis and leakage evaluation results to exclude portions of the tubes from inspection and repair of tube indications is interpreted to constitute a redefinition of the primary to secondary pressure boundary. Inclusion of the permanent alternate repair criteria in TS 6.8.4.j permits deletion of the previous one-time alternate repair criteria for Turkey Point Units 3 and 4 approved by License Amendments 241 and 236. In addition, this amendment request proposes to revise TS 6.9.1.8, Steam Generator Tube Inspection Report, to remove references to the previous one-time alternate repair criteria and provides reporting requirements specific to the permanent alternate repair criteria. The proposed changes to the TS are based on the supporting structural analysis and leakage evaluation completed by Westinghouse Electric Company, LLC. The documentation supporting the Westinghouse analysis is described in Section 4.0 of Attachment 1, including WCAP-17345-P, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3-Loop Model 44F/Model 51F)", Revision 2, June 2011.

Attachment 1 provides the basis for the proposed change, including a detailed description, technical and regulatory evaluations, environmental considerations, and FPL's determination that the proposed change does not involve a significant hazards consideration. The marked-up and proposed TS pages are provided in Enclosures 2 and 3 to Attachment 1.

The license amendments proposed by FPL have been reviewed by the Turkey Point Plant Nuclear Safety Committee. In accordance with 10 CFR 50.91(b)(1), a copy of the proposed license amendment is being forwarded to the State Designee for the State of Florida.

FPL requests approval of the proposed amendment by November 5, 2012 to support the fall 2012 Turkey Point Unit 4 steam generator inspections. Once approved, the amendment shall be implemented prior to entering COLD SHUTDOWN conditions for refueling outage 27.

ADD
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The following commitments, approved by Amendments 241 and 236 and incorporating the H* leak factor proposed in this License Amendment Request, will continue to remain in place, and are also included in Enclosure 5:

1. Turkey Point Units 3 and 4 commits to monitor for tube slippage as part of the steam generator tube inspection program.
2. Turkey Point Units 3 and 4 commits that for the Condition Monitoring Assessment, the component of operational leakage from the prior cycle from below the H* distance will be multiplied by a factor of 1.82 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the Operational Assessment, the difference between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 1.82 and compared to the observed operational leakage. An administrative operational leakage limit will be established to not exceed the calculated value.

Please contact Mr. Robert Tomonto, Licensing Manager, at 305-246-7327 if there are any questions.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on 4/30/2012.

Very truly yours,



Michael W. Kiley
Vice President - Turkey Point Nuclear Generating Station

- Enclosures:
1. Description and assessment of the proposed changes.
 2. Marked-up Technical Specification pages.
 3. Retyped Technical Specification pages.
 4. Marked-up pages for the Technical Specification Bases Control Program.
 5. List of Regulatory Commitments
 6. Response to licensee-specific RAI

cc: Regional Administrator, Region II, USNRC
USNRC Project Manager, Turkey Point
Senior Resident Inspector, USNRC
W. A. Passetti, Florida Department of Health

Turkey Point Nuclear Generating Station Units 3 and 4
Docket Nos.50-250 and 50-251
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ENCLOSURE 1

DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGES

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**Turkey Point Units 3 & 4
License Amendment Request for H*: Alternate Repair Criteria for
Steam Generator Tubesheet Expansion Region**

1.0 INTRODUCTION

Florida Power & Light is proposing to revise the Turkey Point Units 3 and 4 Technical Specifications (TS) 6.8.4.j, "Steam Generator (SG) Program," to permanently exclude portions of the steam generator tubes below the top of the steam generator tubesheet from periodic tube inspections. Application of the supporting structural analysis and leakage evaluation results to exclude portions of the tubes from inspection and repair of tube indications is interpreted to constitute a redefinition of the primary to secondary pressure boundary. Inclusion of the permanent alternate repair criteria in TS 6.8.4.j permits deletion of the previous one-time alternate repair criteria for Turkey Point Units 3 and 4.

In addition, this amendment request proposes to revise TS 6.9.1.8, "Steam Generator Tube Inspection Report," to remove references to the previous temporary alternate repair criteria and provides reporting requirements specific to the permanent alternate repair criteria. The proposed changes to the TS are based on the supporting structural analysis and leakage evaluation completed by Westinghouse Electric Company LLC. The documentation supporting the Westinghouse analysis is described in Section 4.0 and provides the licensing basis for this change.

Table 5-1 of Westinghouse WCAP-17345-P, Revision 2 [Reference 9] provides the 95/95 whole plant H* value of 18.11 inches for plants with Model 44F Steam Generators (Turkey Point Units 3 and 4). In addition, based on WCAP-17091-P, Revision 0 [Reference 2], a leakage factor of 1.82 will be applied for Turkey Point Units 3 and 4.

The NRC previously granted license amendments 241 and 236 [Reference 8] for Turkey Point Units 3 and 4, respectively, to exclude the portion of the tubes below 17.28 inches below the top of the tubesheet on a one-time basis. Under license amendments 241 and 236, tubes with service-induced flaws located greater than 17.28 inches below the top of the tubesheet did not require plugging. Amendment 241 expired at Refueling Outage 26 (RFO-26) for Unit 3 in February, 2012 and Amendment 236 will expire for Unit 4 prior to entering Refueling Outage 27 (RFO-27) in the fall of 2012. (There was no eddy current inspection scheduled for Unit 3 during the RFO-26 outage in February 2012. There is an eddy current inspection scheduled for Unit 4 during the RFO-27 outage in the fall of 2012).

This permanent request for amendment would replace the existing one-time amendments. Approval of this amendment application is requested by November 5, 2012 to support the Turkey Point Unit 4 RFO-27 refueling outage (Fall 2012), since the existing one-time alternate repair criteria approved for Unit 4 Amendment 236 [Reference 8] expires prior to entering RFO-27.

2.0 DETAILED DESCRIPTION OF PROPOSED REVISIONS

TS 6.8.4.j.c. currently states:

- 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 alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

1. For Unit 3 through Refueling Outage 25 and the next operating cycle, and for Unit 4 during Refueling Outage 25 and the subsequent operating cycles until the next scheduled inspection, tubes with service-induced flaws located greater than 17.28 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 17.28 inches below the top of the tubesheet shall be plugged upon detection.

This section would be revised as follows, as noted in bold italic type:

- 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 alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

1. ~~For Unit 3 through Refueling Outage 25 and the next operating cycle, and for Unit 4 during Refueling Outage 25 and the subsequent operating cycles until the next scheduled inspection, tubes~~ Tubes with service-induced flaws located greater than ~~18.11~~ **17.28** 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 ~~18.11~~ **17.28** inches below the top of the tubesheet shall be plugged upon detection.

TS 6.8.4.j.d currently states:

- 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. For Unit 3 through Refueling Outage 25 and the next operating cycle, and for Unit 4 during

Refueling Outage 25 and the subsequent operating cycles until the next scheduled inspection, the portion of the tube below 17.28 inches from the top of the tubesheet is excluded from inspection. 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 tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

This section would be revised as follows, as noted in bold italic type:

- 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. ***For Unit 3 through Refueling Outage 25 and the next operating cycle, and for Unit 4 during Refueling Outage 25 and the subsequent operating cycles until the next scheduled inspection, the*** The portion of the tube below ~~17.28~~ ***18.11*** inches from the top of the tubesheet is excluded from inspection. 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 tube may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

TS 6.9.1.8 currently states:

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, 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.

Note: Report items i, j, and k are applicable following completion of inspections performed through Refueling Outage 25 at Unit 3 (and any inspection performed in the next operating cycle) and Refueling Outage 25 at Unit 4 (and any inspections performed in the subsequent operating cycles until the next scheduled inspection).

- 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 induced leakage rate from the portion of the tubes below 17.28 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and
- k. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

This section would be revised as follows, as noted in bold italic type:

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, 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.

Note: Report items i, j, and k are applicable following completion of inspections performed through Refueling Outage 25 at Unit 3 (and any inspection performed in the next operating cycle) and Refueling Outage 25 at Unit 4 (and any inspections performed in the subsequent operating cycles until the next scheduled inspection).

- 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 induced leakage rate from the portion of the tubes below ~~17.28~~ 18.11 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and
- k. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

3.0 BACKGROUND

Turkey Point Units 3 and 4 are three loop Westinghouse designed plants. Each unit has three replacement Model 44F SGs that were installed in 1982 and 1983 respectively. Each SG has 3214 tubes. Turkey Point Unit 3 (currently shutdown for refueling outage RFO-26) and Unit 4 (currently in cycle 26 operation) have a total of 184 tubes and 64 tubes plugged, respectively. The design of the SG includes Alloy 600 thermally treated tubing, full depth hydraulically expanded tubesheet joints, and stainless steel tube support plates with broached-hole quatrefoils.

The SG inspection scope is governed by TS 6.8.4.j "Steam Generator (SG) Program"; Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" [Reference 3]; EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines" [Reference 4]; EPRI 1019038, "Steam Generator Integrity Assessment Guidelines" [Reference 5]; the Florida Power & Light Steam Generator Integrity Program and the results of the degradation assessments required by the SG Program. Criterion IX, "Control of Special Processes" of 10 CFR Part 50, Appendix B, requires in part that nondestructive testing be accomplished by qualified personnel using qualified procedures in accordance with the applicable criteria. The inspection techniques and equipment are capable of reliably detecting the known and potential specific degradation mechanisms applicable to Turkey Point Units 3 and 4. The inspection techniques, essential variables and equipment are qualified to Appendices H and I, "Performance Demonstration for Eddy Current Examination" of the EPRI Steam Generator Examination Guidelines [Reference 4].

Catawba Nuclear Station, Unit 2 (Catawba), reported indications of cracking following nondestructive eddy current examination of the SG tubes during their fall 2004 outage. In April 2005, NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds" [Reference 6], 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 adjacent to and possibly extending into the weld, also known as tube-to-tubesheet welds, which join the tube to the tubesheet.

The Florida Power & Light Company Steam Generator Integrity Program requires the use of applicable industry operating experience in the operation and maintenance of Turkey Point Units 3 and 4. 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 as discussed in NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections" [Reference 37]. Since the Turkey Point Units 3 and 4 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 Turkey Point Units 3 and 4 to identify tube indications similar to those reported at Catawba within the hot leg tubesheet region if similar inspections are performed at the next scheduled inspection of the SGs.

Potential inspection plans for the tubes and tube welds underwent intensive industry discussions in March 2005. The findings in the Catawba SG tubes present three distinct issues with regard to the SG tubes at Turkey Point Units 3 and 4:

- 1) Indications in internal bulges and over-expansions within the hot leg tubesheet,
- 2) Indications at the elevation of the tack expansion transition, 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 Turkey Point Units 3 and 4 SGs. A validation assessment is also performed to verify that the eddy current techniques utilized are capable of detecting those flaw types that are identified in the degradation assessment. Based on the Catawba operating experience, Turkey Point Units 3 and 4 revised the SG inspection plans for the refueling outage 23 inspections at Unit 3 (fall 2007) and the refueling outage 23 inspections at Unit 4 (fall 2006) to include sampling of bulges (BLB) and over-expansions (OXP) within the tubesheet region on the hot leg side.

The Turkey Point Unit 3 inspection plan for refueling outage 25 in the fall of 2010 included a 50% sample of the hot leg tubesheet to the extent of TTS +3.00 to -17.28 inches, which included a minimum 50% sample of BLG and OXP indications within the TS.

The Turkey Point Unit 4 inspection plan for refueling outage 25 in the fall of 2009 included a 100% of the hot leg tubesheet to the extent of TTS +3.00 to -17.28 inches, which included a minimum 50% sample of BLG and OXP indications within the TS.

The samples in refueling outages 23 and 25 (for both units) were based on the guidance contained in EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7 [Reference 4], and TS 6.8.4.j, "Steam Generator (SG) Program." According to 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). No crack-like indications were detected in either unit during the refueling outage 23 or 25 inspections.

Based on these inspections, no indications of a 360 degree tube sever have been detected in any steam generator at Turkey Point Unit 3 or 4. Consequently, the level of degradation in the Turkey Point steam generators is very limited compared to the

assumption of "all tubes severed" that was utilized in the development of the permanent H* value. Thus, structural integrity will be assured for this permanent alternate repair criteria for the operating period between inspections allowed by TS 6.8.4.j, "Steam Generator (SG) Program".

As a result of these potential issues and the possibility of unnecessarily plugging tubes in the Turkey Point SGs, Florida Power & Light is proposing changes to TS 6.8.4.j to limit the SG tube inspection and repair (plugging) to the safety significant portion of the tubes.

4.0 SUMMARY OF LICENSING BASIS (H* ANALYSIS)

To address the potential impact of the Catawba inspection results, Turkey Point Units 3 and 4 were granted a one-time TS change on November 1, 2006 (Amendments 231 and 226) to limit inspection depth to 17 inches below the top of the hot leg tubesheet [Reference 1]. Reference 1, however, applied only to the inspections conducted during refueling outage 23 (RFO-23) at each unit, which (at the time) were the only inspections conducted since IN 2005-09 was issued.

On July 23, 2009, Turkey Point Units 3 and 4 applied via Letter L-2009-151 [Reference 10] for a permanent alternate repair criteria, known as H*, based on Westinghouse WCAP-17091-P, and proposed changes to Turkey Point Units 3 and 4 TS 6.8.4.j, "Steam Generator (SG) Program" to limit the SG tube inspection and repair (plugging) to the portion of tubing from 17.28 inches below the top of the tubesheet. That amendment also requested to have TS 6.9.1.8, "Steam Generator Tube Inspection Report," revised to provide reporting requirements specific to the permanent alternate repair criteria.

On September 2, 2009, on a teleconference between NRC Staff and industry personnel, the NRC Staff indicated that their concerns with eccentricity of the tube sheet tube bore in normal and accident conditions had not been resolved. Thus, on September 30, 2009, FPL submitted a request (Letter L-2009-209 [Reference 11]) to revise the July 23, 2009 amendment request to be a one-time change applicable for Unit 3 during RFO-25 and the subsequent operating cycles until the next scheduled inspection, and for Unit 4 during RFO-25 and the subsequent operating cycles until the next scheduled inspection.

On October 30, 2009, the NRC granted Turkey Point Units 3 and 4 a one-time TS change (amendments 241 and 236) to limit the inspection depth to 17.28 inches below the top of the hot leg tubesheet [Reference 8]. Reference 8, however, applied only to the inspections conducted during RFO-25 at each unit. Reference 8 expired (for Unit 3) at the end of Unit 3's Cycle-25 in February 2012, and will expire (for Unit 4) at the end of Unit 4's Cycle-26 in the fall of 2012. (There was no eddy current inspection scheduled for Unit 3 during the end of Cycle 25 outage in February 2012. There is an eddy current inspection scheduled for Unit 4 during the end of Cycle 26 outage in the fall of 2012).

On December 29, 2009, (after the NRC issued Amendments 241 and 236 for Turkey Point) the NRC provided a letter [Reference 12] documenting the currently identified and

unresolved issues relating to tubesheet bore eccentricity. The letter contained 14 questions which required resolution before the NRC could complete its review of a permanent amendment request.

The following documents have been prepared by Westinghouse to provide final resolution of the remaining questions identified in the December 29, 2009 NRC letter in support of the permanent H* amendment for Turkey Point Units 3 & 4.

- WCAP-17345-P, Revision 2, "H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3-Loop Model 44F/Model 51F)" [Reference 9].
- 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*" [Reference 13]. This document, which is applicable to Turkey Point's Model 44F SGs, was transmitted to the NRC by Westinghouse letter LTR-NRC-10-68 [Reference 14].
- 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*," [Reference 15] was prepared to support plant determinations of BET measurements and their significant deviation assessment. This document, which is applicable to Turkey Point's Model 44F SGs, was transmitted to the NRC by Westinghouse letter LTR-NRC-10-69 [Reference 16].
- LTR-SGMP-10-33-P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity" [Reference 17]. This document, which is applicable to Turkey Point's Model 44F SGs, was transmitted to the NRC by Westinghouse letter LTR-NRC-10-70 [Reference 18].

Note that the technical information contained in WCAP-17091-P, Revision 0 [Reference 2] remains valid and provides part of the licensing basis for the requested amendment change.

The following table provides the list of the Turkey Point licensing basis documents for H*:

Table 1

Document Number	Revision Number	Title	Reference Number
WCAP-17345-P	2	H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3-Loop Model 44F/Model 51F)	9
WCAP-17091-P	0	H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)	2
LTR-SGMP-09-108-P Attachment	0	Response to NRC Request for Additional Information on H*: Model 44F and Model 51F Steam Generators	19
LTR-SGMP-09-108-P Erratta	****	Errata: Responses to NRC Request for Additional Information on H*: Model 44F and Model 51F Steam Generators	20
LTR-SGMP-10-78-P-Attachment	0	Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to-Tubesheet Contact Pressure and Their Relative Importance to H*	13
LTR-SGMP-10-33-P-Attachment	0	H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity	17

In addition, the following correspondence is also applicable to the Turkey Point permanent alternate repair criteria request:

- A March 28, 2011 letter from the NRC to Southern Nuclear Operating Company [Reference 24] documented the summary of a February 16, 2011 public meeting regarding steam generator tube inspection permanent alternate repair criteria. Enclosure 3 of the NRC letter provided technical NRC Staff questions developed at the meeting. Responses to these questions have been incorporated into WCAP-17345-P, Revision 2 [Reference 9].
- Section 1.3 of Reference 9 identifies revisions in the report to address recommendations from the independent review of the H* analysis performed by MPR Associates. Related to the independent review, a May 26, 2011 letter from the NRC to Southern Nuclear Operating Company [Reference 25] included a presubmittal review request for additional information. The response to the NRC presubmittal review request is provided in Southern Nuclear Operating Company letter NL-11-1178, dated June 20, 2011 [Reference 26].

On July 28, 2011, Virginia Electric and Power Company (Dominion) submitted a license amendment request [Reference 29] for permanent application of the alternate repair

criterion, H*, for Surry Power Station Units 1 and 2. On January 18, 2012, the NRC issued a request to Surry Power Station for additional information [Reference 31]. On February 14, 2012, Surry Power Station submitted its response [Reference 32] to the NRC's Request for Additional Information [Reference 31] related to WCAP-17345-P. The February 14, 2012 Surry responses to the NRC RAIs are applicable to the Surry Westinghouse Model 51F steam generators, and also the Westinghouse Model 44F steam generators installed at Turkey Point. Since the Surry responses to the NRC RAIs also cover the Westinghouse Model 44F steam generators installed at Turkey Point, only one RAI question (RAI No. 15 of Reference 31) requires a plant-specific response for Turkey Point. The Turkey Point response to RAI No. 15 of Reference 31 is provided in Enclosure 6.

5.0 TECHNICAL EVALUATION

To preclude unnecessarily plugging tubes in the Turkey Point Units 3 and 4 SGs, 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 repair methodology is provided in References 2, 9 and 19. 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 Turkey Point Units 3 and 4 Model 44F SGs 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. Table 1 of this enclosure provides the list of documents that provide the technical justification for limiting the inspection in the tubesheet expansion region to less than the full depth of the tubesheet.

The basis for determining 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 pull out and resistance to leakage during plant operation and transients.

The constraint that is provided by the tubesheet precludes tube burst for 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 7] are satisfied by the constraint provided by the tubesheet. Through application of the limited tubesheet inspection scope as described below, the existing operational leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur.

Primary to secondary leakage from tube degradation is assumed to occur in several design basis accidents: steam line break (SLB), 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 Part 100, 10 CFR 50.67, GDC 19). 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. Radiological dose consequences define the limiting accident condition for the H* value. The current licensing basis (based on Reference 33) assumed primary to secondary accident leak rate is 0.20 gallons per minute (gpm) through any one SG, and 0.60 gpm total for each of the SLB, rod ejection and locked rotor events. The 0.20 gpm assumed accident leak rate is 288 gpd, per SG. The SLB leak rate factor for Turkey Point Units 3 and 4 is 1.82 (Table 9-7 in WCAP-17091-P [Reference 2]). Multiplying the TS operational leak rate limit of 150 gpd through any one SG by 1.82 shows that the maximum primary to secondary accident induced leak rate is limited to 273 gpd. This leakage rate is bounded by the current licensing basis assumed primary to secondary accident leak rate of 0.20 gpm (288 gpd) through any one SG for SLB.

The other design basis accidents, such as the postulated locked rotor event and the control rod ejection event, are conservatively modeled using the design specification transients that result in increased temperatures in the SG hot and cold legs for a period of time. As previously noted, dynamic viscosity decreases with increasing temperature. Therefore, leakage would be expected to increase due to decreasing viscosity and increasing differential pressure for the duration of time that there is a rise in RCS temperature. The length of time that a plant with Model 44F SGs 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 for comparison 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, and by a factor of six for the control rod ejection event (less than 10 seconds). This translates into an effective reduction in the leakage factor by the same factor for each event. Therefore, the locked rotor event leak rate factor of 1.66 for Turkey Point Units 3 and 4 is adjusted downward to a factor of 0.83 (Table 9-7, Reference 2). Similarly, the control rod ejection event leak rate factor is reduced by a factor of six, from 2.45 to 0.41 (Table 9-7, Reference 2). Due to the short duration of the transients above NOP 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, SLB is the limiting accident and 1.82 remains the limiting leak rate factor for Turkey Point Units 3 and 4 (Table 9-7 in Reference 2).

It should be noted that some of the discussion in WCAP-17091-P refers to feedline break (FLB) accident analyses. References to FLB analyses, however, are specific to the initial

analyses for other SG models and are not intended to imply that the FLB accident is applicable to Turkey Point Units 3 and 4. Although a limited scope FLB analysis to demonstrate margin to hot leg saturation was performed for Turkey Point Units 3 and 4 in response to NRC questions as part of the Extended Power Uprate review, this analysis is not considered to be part of the design basis for Turkey Point Units 3 and 4.

Plant-specific operating conditions are used to generate the overall leakage factor ratios that are used in the condition monitoring and operational assessments. The plant-specific data provide the initial conditions for application of the transient input data. The results of the analysis of the plant-specific inputs to determine the bounding plant for each model of SG and to assure that the design basis accident contact pressures are greater than the normal operating contact pressure are contained in section 6 of WCAP-17091-P.

As discussed in References 2 and 9, 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 SLB event, a plant cool down event would occur and the subsequent temperature 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 SLB event is 1.82 for Turkey Point Units 3 and 4 (Table 9-7 of WCAP-17091-P).

The leak rate factor of 1.82 for Turkey Point Units 3 and 4 for a postulated SLB has been calculated as shown in Table 9-7 of Reference 2. Turkey Point Units 3 and 4 will apply a factor of 1.82 to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). The leak rate factor of 1.82 in Table 9-7 of Reference 2 applies to both hot and cold legs. Specifically, for the CM assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 1.82 and added to the total leakage from any other source and compared to the assumed accident leak rate. For the OA, the difference between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 1.82 and compared to the observed operational leakage. An administrative operational leakage limit will be established to not exceed the calculated value.

References 2 and 9 redefine 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 over the full range of SG operating conditions, including the most limiting accident conditions. The evaluation in References 2 and 9 determined that degradation in tubing below this safety significant portion of the tube does not require inspection or repair (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.

WCAP-17091-P [Reference 2], section 9.8, provides a review of leak rate susceptibility 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. As a condition of approval of Amendment Numbers 241/236 (Units 3 and 4, respectively) Florida Power & Light committed to monitor for tube slippage as part of the SG tube inspection program. A change was also made to TS 6.9.1.8, "Steam Generator Tube Inspection Report" which added a new reporting requirement for slippage monitoring. Steam Generator tube slippage monitoring conducted during the Unit 3 RFO-25 refueling outage in the fall of 2010, and the Unit 4 RFO-25 refueling outage in the fall of 2009 identified no evidence of slippage. The requirement to monitor for tube slippage will remain in place to support the permanent alternate repair criteria request, and the results of monitoring will be reported in accordance with TS 6.9.1.8.

In addition the NRC staff has requested that licensees determine if there are any significant deviations in the location of the bottom of the expansion transition (BET) relative to the top of tubesheet compared to the assumptions in WCAP-17091-P. Turkey Point Units 3 and 4 has completed the verification of tube expansion locations to determine if any significant deviations exist from the top of tubesheet to the BET. Based on data review and LTR-SGMP-09-111, Rev. 1 [Reference 21] Turkey Point identified that there were 9 tubes in each unit that had significant deviations in the location of the bottom of the expansion transition (BET). Specific details of the BET analysis are documented in Westinghouse documents LTR-SGMP-09-130 [Reference 22] for Turkey Point Unit 4 and LTR-SGMP-10-62 [Reference 23] for Turkey Point Unit 3. The nine tubes identified as having significant deviations (No Tubesheet Expansion or NTE) in Turkey Point Unit 3 were plugged during the EOC-24 refueling outage in the fall of 2010. The nine tubes identified as significant deviations (NTE) in Turkey Point Unit 4 were plugged during the EOC-24 refueling outage in the fall of 2009. One additional tube was plugged in Unit 4 during EOC-24 for having a Hot Leg BET variation >1". Therefore, all tubes with significant deviations in the location of the bottom of the expansion transition (BET) have been removed from service in Turkey Point Units 3 and 4.

6.0 REGULATORY EVALUATION

6.1 Applicable Regulatory Requirements/Criteria

General Design Criteria (GDC) 1, 2, 4, 14, 30, 31, and 32 of 10 CFR 50, Appendix A, define requirements for the reactor coolant pressure boundary (RCPB) with respect to structural and leakage integrity.

GDC 19 of 10 CFR 50, Appendix A, defines requirements for the control room and for the radiation protection of the operators working within it. Accidents involving the leakage or burst of SG tubing comprise a challenge to the habitability of the control room.

10 CFR 50, Appendix B, establishes quality assurance requirements for the design, construction, and operation of safety related components. The pertinent requirements of this appendix apply to all activities affecting the safety related functions of these components. These requirements are described in Criteria IX, XI, and XVI of Appendix B and include control of special processes, inspection, testing, and corrective action.

10 CFR 100, Reactor Site Criteria, establishes reactor site criteria, with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving leakage or tube burst of SG tubing may comprise a challenge to containment and therefore involve an increased risk of radioactive release.

10 CFR 50.67, Accident Source Term, establishes limits on the accident source term used in design basis radiological consequence analyses with regard to radiation exposure to members of the public and to control room occupants.

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, and TS 6.8.4.j.b provides reasonable assurance that the SG tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary. The SG tube performance criteria in NEI 97-06, Revision 3, and TS 6.8.4.j.b are:

- **Structural integrity performance criterion:** All in-service 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. 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 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.

- The primary-to-secondary accident induced leakage rate for any design basis accident, other than 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 0.60 gpm total through all SGs and 0.20 gpm (288 gpd) through any one SG.
- The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day (per TS 3.4.6.2.c).

The safety significant portion of the tube is the length of 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 steam generator operating conditions, including the most limiting accident conditions. The evaluation in this transmittal determined that degradation in tubing below 18.11 inches from the top of the tubesheet portion of the tube does not require plugging and serves as the bases for the SG tube inspection program. As such, the Turkey Point Units 3 and 4 inspection program provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

6.2 No Significant Hazards Consideration

This amendment application proposes to revise TS 6.8.4.j, "Steam Generator (SG) Program," to exclude portions of the tubes within the tubesheet from periodic SG inspections. In addition, this amendment proposes to revise TS 6.9.1.8, "Steam Generator Tube Inspection Report" to provide reporting requirements specific to the permanent alternate repair criteria. Application of the structural analysis and leak rate evaluation results, to exclude portions of the tubes from inspection and repair, is interpreted to constitute a redefinition of the primary to secondary pressure boundary.

The proposed change defines the safety significant portion of the tube that must be inspected and repaired. A justification has been developed by Westinghouse Electric Company LLC in WCAP-17091-P Revision 0 and WCAP-17345-P Revision 2 to identify the specific inspection depth below which any type of axial or circumferential primary water stress corrosion cracking can be shown to have no impact on the performance criteria in Nuclear Energy Institute (NEI) 97-06 [Reference 3], "Steam Generator Program Guidelines," and TS 6.8.4.j.b, "Performance criteria for SG tube integrity."

Florida Power & Light Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment 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 previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the SG inspection and reporting criteria does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed change will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident.

Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed change to the SG tube inspection and repair criteria are the SG tube rupture (SGTR) event and the steam line break (SLB) postulated accident.

Addressing the SGTR event, the required structural integrity margins of the SG 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 presence of the tubesheet and the tube-to-tubesheet joint. Tube burst cannot occur within the thickness of the tubesheet. The tube-to-tubesheet joint 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, and tubesheet rotation. The structural margins against burst, as discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" [Reference 7] and NEI 97-06, "Steam Generator Program Guidelines", [Reference 3] are maintained for both normal and postulated accident conditions.

For the portion of the tube outside of the tubesheet, the proposed change also has no impact on the structural or leakage integrity. Therefore, the proposed change does not result in a significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking below the proposed limited inspection depth is limited by the tube-to-tubesheet crevice. Consequently, negligible normal operating leakage is expected from degradation below the inspected depth within the tubesheet region. The consequences of an SGTR event are not affected by the primary

to secondary leakage flow during the event as primary to secondary leakage flow through a postulated tube that has been pulled out of the tubesheet is essentially equivalent to a tube rupture. Therefore, the proposed change does not result in a significant increase in the consequences of an SGTR. In addition, the selected H* value envelopes the depth within the tubesheet required to prevent a tube pullout.

The probability of a SLB is unaffected by the potential failure of a SG tube as the failure of a tube is not an initiator for a SLB event.

The leak rate factor of 1.82 for Turkey Point Units 3 and 4, for a postulated SLB, has been calculated as shown in References 2, 9 and 19. Turkey Point Units 3 and 4 will apply the factor of 1.82 to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). Through application of the limited tubesheet inspection scope, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. Multiplying the TS operational leak rate limit of 150 gpd (at room temperature) through any one SG by a factor of 1.82 shows that the maximum primary to secondary accident induced leak rate is limited to 273 gpd. This leakage rate is bounded by the current licensing basis assumed primary to secondary accident leak rate of 0.20 gpm (288 gpd) through any one SG for SLB. Since the existing limit on operational leakage continues to ensure that the SLB assumed accident induced leakage will not be exceeded, the consequences of a SLB accident are not increased.

For the CM assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 1.82 and added to the total leakage from any other source and compared to the allowable accident induced leak rate. For the OA, the difference in the leakage between the allowable leakage and the calculated accident induced leakage from sources other than the tubesheet expansion region will be divided by 1.82 and compared to the observed operational leakage.

Based on the above, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change that alters the SG inspection and reporting criteria does not introduce any new equipment, create new failure modes for existing

equipment, or create any new limiting single failures. 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 change involve a significant reduction in a margin of safety?*

Response: No

The proposed change defines the safety significant portion of the tube that must be inspected and repaired. WCAP-17345, Rev. 2 [Reference 9] identifies the specific inspection depth below which any type of tube degradation is shown to have no impact on the performance criteria in NEI 97-06 Rev. 3, "Steam Generator Program Guidelines" [Reference 3] and TS 6.8.4.j, "Steam Generator (SG) Program."

The proposed change that alters the SG inspection and reporting criteria maintains the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute 97-06, "Steam Generator Program Guidelines" [Reference 3], and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" [Reference 7], are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor Coolant Pressure Boundary," GDC 15, "Reactor Coolant System Design," GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and GDC 32, "Inspection of Reactor Coolant Pressure Boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation, the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse WCAP-17091-P, Rev. 0 [Reference 2] and WCAP-17345, Rev. 2 [Reference 9] 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 SLB leak rate factor for Turkey Point Units 3 and 4 is 1.82 (Table 9-7 in WCAP-17091-P). Multiplying the TS operational leak rate limit of 150 gpd

through any one SG by the leak rate factor of 1.82 shows that the maximum primary to secondary accident induced leak rate is limited to 273 gpd. This leakage rate is bounded by the current licensing basis assumed primary to secondary accident leak rate of 0.20 gpm (288 gpd) through any one SG for SLB.

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

6.3 Precedents

The proposed change to Turkey Point Technical Specification (TS) 6.8.4.j and 6.9.1.8 is similar to the following proposed changes, which have been submitted to revise TS for permanent alternate repair criteria:

- 6.3.1 Virginia Electric and Power Company (Dominion) Surry Power Station Units 1 and 2 License Amendment Request Permanent Alternate Repair Criteria For Steam Generator Tube Inspection and Repair, dated July 28, 2011, [Reference 29] (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 36]).
- 6.3.2 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 30] (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 34]).

6.4 Conclusion

The safety significant portion of the tube is the length of tube that is engaged within the tubesheet to the top of the tubesheet (secondary face) that is required to maintain structural and leakage integrity over the full range of steam generating operating conditions, including the most limiting accident conditions. The H* Analysis determined that degradation in tubing below the safety significant portion of the tube does not require plugging and serves as the basis for the limited tubesheet inspection criteria, which are intended to ensure the primary to secondary leak rate during any accident does not exceed the leak rate assumed in

the accident analysis. Based on the considerations 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.

7.0 ENVIRONMENTAL CONSIDERATIONS

Turkey Point Unit 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 amendments meet 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.

8.0 REFERENCES

1. NRC letter, "Turkey Point Plant, Units 3 and 4 – Issuance of Amendments Regarding Steam Generator Alternate Repair Criteria (TAC Nos. MD 1380 and MD 1381).," dated November 1, 2006. (ADAMS Accession # ML062990169).
2. Westinghouse Electric Company WCAP-17091-P, Rev 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)" Revision 1, dated June, 2009. (Submitted to NRC as Enclosure 6 of Reference 10).
3. NEI 97-06, "Steam Generator - Program Guidelines," Revision 3, January 2011.
4. EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7.
5. EPRI 1019038, "Steam Generator Integrity Assessment Guidelines," November 2009.
6. NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," April 7, 2005.
7. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976.
8. NRC letter, "Turkey Point Units 3 and 4 – Issuance of Amendments Regarding H*: Alternate Repair Criteria for Steam Generator Tubesheet Expansion Region (TAC

Nos. ME1754 and ME1755),” dated October 30, 2009. (ADAMS Accession # ML092990489).

9. Westinghouse Electric Company WCAP-17345-P, Rev 2 “H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3-Loop Model 44F/Model 51F)”. (Submitted to NRC July 28, 2011 as Attachment 4 of Reference 29).
10. Florida Power & Light to the NRC (Letter L-2009-151): “Turkey Point Units 3 and 4 Docket Nos. 50-250 and 50-251 License Amendment Request No. 197 for H*: Alternate Repair Criteria for Steam Generator Tubesheet Expansion Region” dated July 23, 2009. (ADAMS Accession # ML092300059).
11. Florida Power & Light to the NRC (Letter L-2009-209): “Turkey Point Units 3 and 4 Docket Nos. 50-250 and 50-251 Response to Request for Additional Information Regarding License Amendment Request for H*: Alternate Repair Criteria for Steam Generator Tubesheet Expansion Region” dated September 30, 2009. (ADAMS Accession # ML092800222).
12. NRC Letter, “Turkey Point Nuclear Plant, Units 3 and 4 –Request for Additional Information Regarding the Permanent Alternate Repair Criteria License Amendment Request (TAC Nos. ME1754 and ME1755),” dated December 29, 2009. (ADAMS Accession # ML093561371).
13. Westinghouse 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*".
14. Westinghouse letter LTR-NRC-10-68, “Submittal of LTR-SGMP-10-78-P- Attachment and LTR-SGMP-10-78 NP-Attachment, "Effects of Tubesheet Bore Eccentricity and Dilation on Tube-to- Tubesheet Contact Pressure and Their Relative Importance to H*," (Proprietary/Non-Proprietary) for Review and Approval”, dated November 9, 2010.
15. Westinghouse 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*".
16. Westinghouse letter LTR-NRC-10-69, “Submittal of LTR-SGMP-09-111-P- Attachment, Rev. 1 and LTR-SGMP-09-111 NP-Attachment, Rev. 1, "Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*," (Proprietary/Non-Proprietary) for Review and Approval”, dated November 10, 2010. (ADAMS Accession # ML103400083).
17. Westinghouse LTR-SGMP-10-33-P-Attachment, "H* Response to NRC Questions Regarding Tubesheet Bore Eccentricity".
18. Westinghouse letter LTR-NRC-10-70, “Submittal of LTR-SGMP-10-33-P- Attachment and LTR-SGMP-10-33 NP-Attachment, "H*: Response to NRC Questions Regarding Tubesheet Bore Eccentricity," (Proprietary/Non-Proprietary) for Review and Approval”, dated November 11, 2010.

19. Westinghouse Letter LTR-SGMP-09-108-P – Attachment “Response to NRC Request for Additional Information on H*; Model 44F and Model 51F Steam Generators”, dated August 27, 2009.
20. Westinghouse Letter LTR-SGMP-09-108-P – Errata, “Errata: Responses to NRC Request for Additional Information on H*; Model 44F and Model 51F Steam Generators”, dated September 8, 2009.
21. Westinghouse letter LTR-SGMP-09-111-P-Attachment “Acceptable Value of the Location of the Bottom of the Expansion Transition (BET) for Implementation of H*” Rev 1, dated September 2010.
22. Westinghouse letter LTR-SGMP-09-130, “Turkey Point Unit 4: Position of the Bottom of the Tubesheet Expansion Transition”, dated September 30, 2009.
23. Westinghouse letter LTR-SGMP-10-62, Rev 1 “Turkey Point Unit 3: Position of the Bottom of the Tubesheet Expansion Transition”, dated Oct 7, 2011.
24. NRC to Southern Nuclear Operating Company, Inc. letter, Vogtle Electric Generating Plant Units 1 and 2 - Summary of February 16, 2011 Meeting with Southern Nuclear Operating Company, Inc. and Westinghouse on Technical Issues Regarding Steam Generator Tube Inspection Permanent Alternate Repair Criteria, dated March 28, 2011. (ADAMS Accession No. ML110660648).
25. NRC to Southern Nuclear Operating Company, Inc. letter, Vogtle Electric Generating Plant Units 1 and 2 - Presubmittal Consideration of Steam Generator Alternative Repair Criteria Requirements Request for Additional Information, dated May 26; 2011 (ADAMS Accession No. ML11140A099).
26. Southern Nuclear Operating Company, Inc. to NRC letter NL-11-1178, Vogtle Electric Generating Plant - Response to Presubmittal Consideration of Steam Generator Alternative Repair Criteria Requirements Request for Additional Information, dated June 20, 2011.
27. Westinghouse Electric Company WCAP-17345-NP, Rev 2 “H*: Resolution of NRC Technical Issue Regarding Tubesheet Bore Eccentricity (3-Loop Model 44F/Model 51F)”. (Submitted to NRC July 28, 2011 as Attachment 5 of Reference 29).
28. Westinghouse Electric Company WCAP-17091-NP, Rev 0, “H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)” Revision 1, dated June, 2009. (Submitted to NRC as Enclosure 5 of Reference 10).
29. “Virginia Electric and Power Company (Dominion) Surry Power Station Units 1 and 2 License Amendment Request Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair”, dated July 28, 2011. (ADAMS Accession No. ML11215A058).
30. 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. (ADAMS Accession No. ML11188A107).
31. NRC letter to Surry Power Station. "Surry Power Station, Units Nos. 1 and 2- Request for Additional Information Regarding the Steam Generator License Amendment Request to Revise Technical Specification for Permanent Alternate Repair Criteria (TAC Nos. ME6803 and ME6804)", dated January 18, 2012. (ADAMS Accession No. ML12006A001).
 32. Surry Power Station letter to NRC, "Virginia Electric and Power Company (Dominion) Surry Power Stations 1 and 2 Response to Request for Additional Information Related to License Amendment Request for Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair, dated February 14, 2012. (ADAMS Accession No. ML12048A676).
 33. NRC Letter dated June 23, 2011, Turkey Point Units 3 and 4 - Issuance of Amendments Regarding Alternative Source Term (TAC Nos. ME1624 AND ME1625) (ADAMS Accession No. ML110800666).
 34. NRC Letter to Catawba Nuclear Station, "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.
 35. Florida Power & Light to the NRC: (Letter L-2011-438): "Turkey Point Units 3 and 4 Docket Nos. 50-250 and 50-251 Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request No. 205" dated October 15, 2011. (ADAMS Accession No. ML11292A032).
 36. 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)", dated April 17, 2012 (ADAMS Accession No. ML12109A270).
 37. NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," August 30, 2004.

ENCLOSURE 2

Marked-up Technical Specification (TS) Pages and Inserts

Refer to the attached markup of the TS showing the proposed changes. The attached markups reflect the currently issued version of the TS and Facility Operating License. At the time of submittal, the Facility Operating License was revised through License Amendments 248 and 244 to Renewed Facility Operating Licenses DPR-31 for Turkey Point Unit 3 and DPR-41 for Turkey Point Unit 4.

The following table lists license amendment requests that are awaiting NRC approval and may impact the currently issued version of the Facility Operating License affected by this LAR.

LAR Number	Technical Specification Section Affected
None	

The following TS pages are included in the enclosed markup:

TS Section	Title	Page
6.8.4.j.c	STEAM GENERATOR (SG) PROGRAM	6-18a
6.8.4.j.d	STEAM GENERATOR (SG) PROGRAM	6-18b
6.9.1.8	STEAM GENERATOR TUBE INSPECTION REPORT	6-22a

(3 pages follow)

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- 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 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. 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 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 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 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.
 3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System 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 alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

1. ~~For Unit 3 through Refueling Outage 25 and the next operating cycle, and for Unit 4 during Refueling Outage 25 and the subsequent operating cycles until the next scheduled inspection,~~ Tubes with service-induced flaws located greater than ~~17.28~~ 18.11 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 ~~17.28~~ 18.11 inches below the top of the tubesheet shall be plugged upon detection.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- 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. ~~For Unit 3 through Refueling Outage 25 and the next operating cycle, and for Unit 4 during Refueling Outage 25 and the subsequent operating cycles until the next scheduled inspection,~~ The portion of the tube below ~~18.11~~ 17.28 inches from the top of the tubesheet is excluded from inspection. 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 tube 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 outages 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 portion of a SG tube not excluded above, 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-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-secondary leakage.

k. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.
- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

ADMINISTRATIVE CONTROLS

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, 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.

~~Note: Report items i, j, and k are applicable following completion of inspections performed through Refueling Outage 25 at Unit 3 (and any inspection performed in the next operating cycle) and Refueling Outage 25 at Unit 4 (and any inspections performed in the subsequent operating cycles until the next scheduled inspection).~~

- 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 induced leakage rate from the portion of the tubes below 18.1147.28 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and
- k. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

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ENCLOSURE 3

RETYPE TS PAGES WITH THE PROPOSED CHANGES INCORPORATED

(3 pages follow)

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- 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 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. 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 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 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 0.60 gpm total through all SGs and 0.20 gpm through any one SG at room temperature conditions.
 3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System 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 alternate tube repair criteria shall be applied as an alternative to the 40% depth based criteria:

1. Tubes with service-induced flaws located greater than 18.11 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 18.11 inches below the top of the tubesheet shall be plugged upon detection.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- 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 portion of the tube below 18.11 inches from the top of the tubesheet is excluded from inspection. 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 tube 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 outages 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 portion of a SG tube not excluded above, 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-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-secondary leakage.

k. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

- a. The definition of the CRE and the CRE boundary.
- b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

ADMINISTRATIVE CONTROLS

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, 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.
- 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 induced leakage rate from the portion of the tubes below 18.11 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 1.82 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined, and
- k. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

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ENCLOSURE 4

Marked-up pages for the Technical Specification Bases Control Program

Turkey Point Units 3 and 4 Administrative Procedure
0-ADM-536, "Technical Specification Bases Control Program"
Page 57

(1 page follows)

ATTACHMENT 1

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TECHNICAL SPECIFICATION BASES

3/4.4.5 (Cont'd)Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall from **18.11** ~~17.28~~ inches below the top of the tubesheet on the hot leg side to **18.11** ~~17.28~~ inches below the top of the tubesheet on the cold leg side. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.j and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, the gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation. Tube collapse is defined as, for the load displacement curve for a given structure, collapse occurs at the top of the load verses displacement curve where the slope of the curve becomes zero. The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term significant is defined as an accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse to be established. For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

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ENCLOSURE 5

LIST OF REGULATORY COMMITMENTS FOR THIS REQUEST

**Turkey Point Units 3 & 4
License Amendment Request for Permanent Alternate Repair Criteria (H*) for
Steam Generator Tubesheet Expansion Region**

The following commitments, approved by Amendments 241 and 236 and incorporating the H* leak factor proposed in this License Amendment Request, will continue to remain in place (these are not new commitments):

REGULATORY COMMITMENTS		IMPLEMENT
1	Turkey Point Units 3 and 4 commit to monitor for tube slippage as part of the steam generator tube inspection program.	At each scheduled inspection required by the Steam Generator Program
2	Turkey Point Units 3 and 4 commit to the following: Turkey Point Units 3 and 4 will apply a factor of 1.82 to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). Specifically, for the CM assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 1.82 and added to the total leakage from any other source and compared to the assumed accident induced leak rate. For the OA, the difference between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 1.82 and compared to the observed operational leakage. An administrative operational leakage limit will be established to not exceed the calculated value.	An administrative operational leakage limit associated with the Permanent Alternate Repair Criteria (PARC) and the 1.82 leakage factor will be established in the CMOA starting with the Unit 4 Fall 2012 RFO and the Unit 3 Fall 2013 RFO.

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ENCLOSURE 6

RESPONSE TO LICENSEE-SPECIFIC RAI

RESPONSE TO LICENSEE-SPECIFIC RAI

The February 14, 2012 Surry responses [Reference 1] to the NRC RAIs [Reference 2] are applicable to the Surry Westinghouse Model 51F steam generators, and also the Westinghouse Model 44F steam generators installed at Turkey Point.

Since the Surry responses to the NRC RAIs also cover the Westinghouse Model 44F steam generators installed at Turkey Point, only one RAI question (RAI No. 15 of Reference 2) requires a plant-specific response for Turkey Point.

The following is RAI No. 15 from Reference 2.

RAI No. 15:

Verify that regulatory commitments pertaining to monitoring for tube slippage and for primary to secondary leakage, as described in Dominion letter dated December 16, 2010 (NRC ADAMS Accession No. ML103550206), Attachment 1, page 10 of 23, remain in place. In addition, revise the proposed amendment to include a revision to technical specification limit on primary to secondary leakage from 150 gallons per day (gpd) to 83 gpd (150 divided by the proposed 1.8 leakage factor), or provide a regulatory basis for not making this change.

Response:

The regulatory commitments pertaining to monitoring for tube slippage and for primary to secondary leakage as described in Florida Power & Light Letter L-2009-209 [Reference 3] remain in place. Turkey Point is not proposing any changes to the primary to secondary LEAKAGE limit as specified in 3/4.4.6 "REACTOR COOLANT SYSTEM LEAKAGE" based on the following:

Primary to secondary leakage from tube degradation is assumed to occur in several design basis accidents: steam line break (SLB), 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 Part 100, 10 CFR 50.67, GDC 19). 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. Radiological dose consequences define the limiting accident condition for the H* value.

The constraint that is provided by the tubesheet precludes tube burst for cracks within the tubesheet. The criteria for tube burst described in NEI 97-06 [Reference 4] and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," [Reference 5] are satisfied by the constraint provided by the tubesheet. Through application of the limited tubesheet inspection scope as described below, the existing operational leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur.

The current licensing basis (based on Reference 6) assumed primary to secondary accident leak rate is 0.20 gpm (288 gpd) through any one SG, and 0.60 gpm total for each of the SLB, rod ejection and locked rotor events.

The maximum leak rate factor for Turkey Point Units 3 and 4 is 1.82 (Table 9-7 in WCAP- 17091-P [Reference 7]) for the SLB event. Multiplying the TS operational leak rate limit of 150 gpd through any one SG by 1.82 shows that the maximum primary to secondary accident induced leak rate is limited to 273 gpd. This leakage rate is bounded by the current licensing basis assumed primary to secondary accident leak rate of 0.20 gpm (288 gpd) through any one SG for SLB.

Therefore, because the maximum induced leakage rate (considering the H* leak rate factor) is bounded by the current licensing basis assumed primary to secondary accident induced leak rate, the technical specification operational leak rate limit of 150 gpd is not required to be revised.

References:

1. Surry Power Station letter to NRC, "Virginia Electric and Power Company (Dominion) Surry Power Stations 1 and 2 Response to Request for Additional Information Related to License Amendment Request for Permanent Alternate Repair Criteria for Steam Generator Tube Inspection and Repair, dated February 14, 2012. (ADAMS Accession No. ML12048A676).
2. NRC letter to Surry Power Station. "Surry Power Station, Units Nos. 1 and 2- Request for Additional Information Regarding the Steam Generator License Amendment Request to Revise Technical Specification for Permanent Alternate Repair Criteria (TAC Nos. ME6803 and ME6804)", dated January 18, 2012. (ADAMS Accession No. ML12006A001).
3. Florida Power & Light to the NRC (Letter L-2009-209): "Turkey Point Units 3 and 4 Docket Nos. 50-250 and 50-251 Response to Request for Additional Information Regarding License Amendment Request for H*: Alternate Repair Criteria for Steam Generator Tubesheet Expansion Region" dated September 30, 2009. (ADAMS Accession # ML092800222).
4. NEI 97-06, "Steam Generator - Program Guidelines," Revision 3, January 2011.
5. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976.
6. NRC Letter dated June 23, 2011, Turkey Point Units 3 and 4 - Issuance of Amendments Regarding Alternative Source Term (TAC Nos. ME1624 AND ME1625) (ADAMS Accession No. ML110800666).
7. Westinghouse Electric Company WCAP-17091-P, Rev 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model 44F)" Revision 1, dated June, 2009.