NEXTera ENERGY SEABROOK

May 28, 2009

SBK-L-09118
Docket No. 50-443

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

Seabrook Station

License Amendment Request 09-03
Revision to Technical Specification 6.7.6.k, "Steam Generator (SG) Program," for Permanent Alternate Repair Criteria (H*)

Reference: NextEra Energy Seabrook, LLC letter SBK-L-09108, Submittal of Westinghouse Topical Report WCAP-17071 H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F), May 5, 2009

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), NextEra Energy Seabrook, LLC (NextEra) is submitting License Amendment Request (LAR) 09-03 for an amendment to Operating License NPF-86 for Seabrook Station.

This LAR proposes a change to Technical Specification (TS) 6.7.6.k, Steam Generator (SG) Program, to exclude a portion of the tubes below the top of the steam generator tube sheet from periodic steam generator tube inspections. The change also adds additional reporting criteria to TS 6.8.1.7, Steam Generator Tube Inspection Report. WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," April 2009, supports the proposed change. At the request of the NRC staff, the referenced letter provided WCAP-17071-P to the staff in advance of this LAR.

Attachment 1 to this letter provides NextEra's evaluation of the proposed change. Attachment 2 contains mark-ups of the technical specifications showing the requested changes. Attachment 3, which contains a proposed TS bases change, is provided for information only. The TS bases will be revised in accordance with TS 6.7.6.j, Technical Specification (TS) Bases Control Program, upon implementation of the license amendment. Enclosed in attachments 4 and 5 are copies of WCAP-17071-P, H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F) (Proprietary), and WCAP-17071-NP, H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)

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(Non-proprietary), respectively. Enclosed in attachment 6 is Westinghouse authorization letter CAW-09-2565 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

Attachment 4 contains information proprietary to Westinghouse Electric Company LLC, and it is supported by an affidavit in attachment 6 signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b) (4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information that is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-09-2565 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

As discussed in the enclosed LAR, the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92. A copy of this letter and the enclosed LAR has been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b). NextEra has determined that LAR 09-03 meets the criteria of 10 CFR 51.22(b) for a categorical exclusion from the requirements for an Environmental Impact Statement. The Station Operation Review Committee has reviewed this LAR.

NextEra requests approval of the proposed license amendment by October 1, 2009 with a 30-day implementation period to support implementation of the amendment during the fall 2009 refueling outage.

This letter makes the following commitments, which are included in attachment 7:

1. NextEra Energy Seabrook, LLC commits to perform a one-time verification of tube expansion locations to determine if any significant deviations exist from the top of tubesheet to the bottom of expansion transition (BET). If any significant deviations are found, the condition will be entered into the plant's corrective action program and dispositioned.

2. NextEra Energy Seabrook, LLC commits to monitor for tube slippage as part of the steam generator tube inspection program.

Should you have any questions regarding this letter, please contact Mr. Michael O'Keefe, Licensing Manager, at (603) 773-7745.

Very truly yours,

NextEra Energy Seabrook, LLC

Gene F. St. Pierre
Vice President - North

Attachments:

1. NextEra Energy Seabrook, LLC's Evaluation of the Proposed Change
2. Mark-up of the Technical Specifications
3. Proposed TS Bases Change
4. WCAP-17071 (Proprietary)
5. WCAP-17071 (Non-proprietary)
6. Westinghouse Letter CAW-09-2565
7. List of Regulatory Commitments

cc: S. J. Collins, NRC Region I Administrator
D. L. Egan, NRC Project Manager, Project Directorate I-2
W. J. Raymond, NRC Senior Resident Inspector

Mr. Christopher M. Pope, Director Homeland Security and Emergency Management
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NEXTera ENERGY SEABROOK

AFFIDAVIT

SEABROOK STATION UNIT 1

Facility Operating License NPF-86

Docket No. 50-443

License Amendment Request 09-03

Revision to Technical Specification 6.7.6.k, Steam Generator (SG) Program, for Permanent Alternate Repair Criteria (H*)

The following information is enclosed in support of this License Amendment Request:

- **Attachment 1** **NextEra Energy Seabrook, LLC's Evaluation of the Proposed Change**
- **Attachment 2** **Mark-up of the Technical Specifications**
- **Attachment 3** **Proposed TS Bases Change**
- **Attachment 4** **WCAP-17071 (Proprietary)**
- **Attachment 5** **WCAP-17071 (Non-proprietary)**
- **Attachment 6** **Westinghouse Letter CAW-09-2565**
- **Attachment 7** **List of Regulatory Commitments**

I, Gene St. Pierre, Site Vice President of NextEra Energy Seabrook, LLC hereby affirm that the information and statements contained within this License Amendment Request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed before me this 28th day of May, 2009

Shirley Sweeney

Notary Public

SHIRLEY SWEENEY MY COMMISSION EXPIRES OCT 6, 2009 NOTARY PUBLIC NEW HAMPSHIRE

Gene St. Pierre

Gene St. Pierre
Vice President - North

Attachment 1

NextEra Energy Seabrook, LLC's Evaluation of the Proposed Change

Subject: License Amendment Request 09-03, Revision to Technical Specification 6.7.6.k, Steam Generator (SG) Program, for Permanent Alternate Repair Criteria (H*)

1.0 SUMMARY DESCRIPTION

NextEra Energy Seabrook, LLC (NextEra) proposes to revise Technical Specification (TS) 6.7.6.k, Steam Generator (SG) Program, to exclude portions of the steam generator tube below the top of the SG tubesheet from periodic steam generator 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. The NRC previously granted NextEra a similar, one-time amendment to exclude the portion of the tubes below 17 inches from the top of the tubesheet [Reference 1]. This permanent change is supported by Westinghouse Electric Company, LLC WCAP-17071-P, H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F), April, 2009 [Reference 2]. WCAP-17071-P recommends the 95/50 H* value of 11.2 inches; however, NextEra has chosen to use an H* value of 13.1 inches for additional conservatism. This more conservative value was discussed between the NRC staff and industry representatives on April 24, 2009 and May 1, 2009.

The existing one-time amendment expires at the end of the current operating cycle. Therefore, NextEra requests approval of the proposed license amendment by October 1, 2009 with a 30-day implementation period to support implementation of the amendment during the fall 2009 refueling outage.

2.0 DETAILED DESCRIPTION

<u>Proposed changes to the current TSs</u>:

TS 6.7.6.k.c. is revised as follows (new text in bold italics and deleted text in strikethrough):

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 ~~may~~ ***shall*** be applied as an alternative to the 40% depth based criteria:

   ~~1. During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, flaws identified in the portion~~

~~of the tube below 17 inches from the top of the hot leg do not require plugging.~~

~~During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection all tubes with flaws identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the tubesheet shall be plugged.~~

Tubes with service-induced flaws located greater than 13.1 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 13.1 inches below the top of the tubesheet shall be plugged upon detection.

TS 6.7.6.k.d is revised as follows:

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 13.1 inches below the top of the tubesheet on the hot leg side to 13.1 inches below the top of the tubesheet on the cold leg side,*** ~~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. ~~During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded from inspection when the alternate tube repair criteria in TS 6.7.6.k.c are implemented.~~ The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

TS 6.7.6.k.d.3 is revised as follows:

If crack indications are found in any SG tube, ***from 13.1 inches below the top of the tubesheet on the hot leg side to 13.1 inches below the top of the tubesheet on the cold leg side***, 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 nondestructive 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.

TS 6.8.1.7 is revised to include three additional reporting requirements, i, j, and k:

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 6.7.6.k, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

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

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

j. The calculated accident induced leakage rate from the portion of the tubes below 13.1 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 2.03 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 TECHNICAL EVALUATION

3.1 Background

Seabrook Station is a four loop Westinghouse designed plant with Model F steam generators having 5626 tubes in each SG. A total of 161 tubes are currently plugged in all four SG. 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 steam generator inspection scope is governed by TS 6.7.6.k, Steam Generator (SG) Program; Nuclear Energy Institute (NEI) 97-06, Steam Generator Program Guidelines [Reference 3]; EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines [Reference 4]; EPRI 1012987, Steam Generator Integrity Assessment Guidelines [Reference 5]; Seabrook Station "Steam Generator Management Reference Manual;" 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 Seabrook Station. The inspection techniques, essential variables and equipment are qualified to Appendix H, "Performance Demonstration for Eddy Current Examination" of the EPRI Steam Generator Examination Guidelines.

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

NextEra policies and programs, as well as TS 6.7.6.k, require the use of applicable industry operating experience in the operation and maintenance of Seabrook Station. The recent 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 steam generators) with techniques capable of finding potential forms of degradation that may be occurring at these locations (as discussed in Generic Letter 2004-001, "Requirements for Steam Generator Tube Inspections"). Since the Seabrook Station Westinghouse Model F steam generators were fabricated with Alloy 600 thermally treated tubes similar to the Catawba Unit 2 Westinghouse Model D5 steam generators, a potential exists for Seabrook Station to identify tube indications similar to those reported at Catawba within the hot leg tubesheet region if similar inspections are performed during the fall 2009 refueling outage.

Potential inspection plans for the tubes and tube welds underwent intensive industry discussions in March 2005. The findings in the Catawba SG tubes present two distinct issues with regard to the SG tubes at Seabrook Station:

1) Indications in internal bulges and over expansions within the hot leg tubesheet, and

2) Indications at the elevation of the tack expansion transition.

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 Seabrook Station 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, Seabrook Station revised the SG inspection plan for the fall 2006 refueling outage (OR11) to include sampling of bulges and over expansions within the tubesheet region on the hot leg side. The sample was based on the guidance contained in EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7, and TS 6.7.6.k, 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). However, degradation was not detected in the tubesheet region in OR11.

As a result of these potential issues and the possibility of unnecessarily plugging tubes in the Seabrook Station SGs, NextEra is proposing changes to TS 6.7.6.k to limit the steam generator tube inspection and repair (plugging) to the portion of tubing from 13.1 inches below the top of the tubesheet.

3.2 Evaluation

To preclude unnecessarily plugging tubes in the Seabrook Station SG, tube inspections will be limited to identifying and plugging degradation in the portion of the tube within the tubesheet necessary to maintain structural and leakage integrity in both normal and accident conditions. The technical evaluation for the inspection and repair methodology is provided in Westinghouse Electric Company, LLC WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)". The 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 Seabrook Station Model F SG 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 pull out from the tubesheet due to axial end cap loads acting on the tube and to ensure that the accident induced leakage limits are not exceeded. WCAP-17071-P provides 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 portion of the tube that requires eddy current inspection 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 WCAP-17071-P. The tube-to-tubesheet radial contact pressure provides resistance to tube pull out.

Primary-to-secondary leakage from tube degradation in the tubesheet area is assumed to occur in several design basis accidents: feedwater line break (FLB), 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. 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. During normal operation, the TS limit primary-to-secondary leakage to 150 gpd through any one SG. In addition, leakage is limited by a leakage factor of 2.03. For Seabrook Station, this factor limits operational leakage to 246 gpd during normal operation to ensure that the assumed accident induced leakage of 500 gpd in the faulted SG is not exceeded under accident conditions. The limiting leakage ratio of 2.03 is independent of the H* distance defined in WCAP-17071-P.

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 and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," [Reference 7] are satisfied due to the constraint provided by the tubesheet. Through application of the limited tubesheet inspection scope as described below, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur. The assumed accident induced leak rate limit is 500 gallons per day (gpd) through the faulted steam generator and 940 gpd for the remaining three SGs for SLB. Based upon the limiting leak rate factor of 2.03, an operational leak rate of less than 246 gpd would be required to prevent exceeding the assumed accident induced leak rate limit of 500 gpd. Therefore, the technical specification leak rate limit of 150 gpd provides significant added margin against the 500 gpd accident analysis leak rate assumption.

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 pressure contact pressure are contained in section 6 of WCAP-17071-P.

The leak rate ratio (accident induced leak rate to operational leak rate) is directly proportional to the change in differential pressure and inversely proportional to the dynamic viscosity. Since dynamic viscosity decreases with an increase in temperature, an increase in temperature results in an increase in leak rate. However, for both the postulated SLB and FLB events, a plant cool down event would occur and the subsequent temperatures in the reactor coolant system (RCS) would not be expected to exceed the temperatures at plant no load conditions. Thus, an increase in leakage would not be expected to occur as a result of the temperature change. 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 and FLB events is 2.02 for Seabrook, and 2.03 is the bounding value for all steam generator designs.

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. For transients other than a SLB and FLB, the length of time that a plant with model F SGs will exceed the normal operating differential pressure across the tubesheet is less than 30 seconds. As the accident induced leakage performance criteria is defined in gallons per minute, the leak rate for a locked rotor 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 because of the short duration (less than 30 seconds) of the elevated pressure differential. This translates into an effective reduction in the leakage factor by the same factor of two for the locked rotor event. Therefore, for the locked rotor event, the leakage factor of 1.74 (Table 9-7, Reference 2) for Seabrook is adjusted downward to a factor of 0.87. Similarly, for the control rod ejection event, the duration of the elevated pressure differential is less than 10 seconds. Thus, the peak leakage factor is reduced by a factor of six, from 2.65 to 0.44. Due to the short duration of the transients above normal operating pressure differential, no leakage factor is required for the locked rotor and control rod ejection events (i.e., the leakage factor is under 1.0 for both transients).

The plant transient response following a full power double-ended main feedwater line rupture corresponding to "best estimate" initial conditions and operating characteristics, as generally presented in steam generator design transients and in the UFSAR Chapter 15.0 safety analysis, indicates that the transient for a Model F SG exhibits a cool down characteristic instead of a heat-up transient. The use of either the component design specification transient or the Chapter 15.0 safety transient for leakage analysis for FLB is overly conservative because:

- The assumptions on which the FLB design transient is based are specifically intended to establish a conservative structural (fatigue) design basis for RCS components; however, H* does not involve component structural and fatigue issues. The best estimate transient is considered more appropriate for use in the H* leakage calculations.

  For the Model F SG, the FLB transient curve (Figure 9-5, Reference 2) represents a double-ended rupture of the main feedwater line concurrent with both loss of offsite power (loss of main feedwater and reactor coolant pump coast down) and turbine trip.

- The assumptions on which the FLB safety analysis is based are specifically intended to establish a conservative basis for minimum auxiliary feedwater (AFW) capacity and combine worst case assumptions, which are exceptionally more severe when the FLB occurs inside containment. For example, environmental errors that are applied to reactor trip and engineered safety features actuation would be less severe. This would result in much earlier reactor trip and greatly increase the SG liquid mass available to provide cooling to the RCS.

A SLB event would have similarities to a FLB except that the break flow path would include the secondary separators, which could only result in an increased initial cooldown (because of retained liquid inventory available for cooling) when compared to the FLB transient. A SLB could not result in more limiting RCS temperature conditions than a FLB.

In accordance with plant operating procedures, the operator would take action following a high energy secondary line break to stabilize the RCS conditions. The expectation for a SLB or FLB with credited operator action is to stop the system cooldown through isolation of the faulted steam generator and control of temperature by the AFW system. Steam pressure control would be established by either the steam generator safety valves or the atmospheric relief valves. For any of the steam pressure control operations, the maximum RCS temperature would be approximately the no load temperature and would be well below normal operating temperature.

Since the best estimate FLB transient temperature would not be expected to exceed the normal operating temperature, the viscosity ratio for the FLB transient is set to 1.0.

The leakage factor of 2.02 for Seabrook Station for a postulated SLB/FLB has been calculated as shown in Table 9-7 of Reference 2. However, NextEra will apply a factor of 2.03 to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). The leakage factor of 2.03 is a bounding value for all steam generators, both hot and cold legs, in Table 9-7 of Reference 2. 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 2.03 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. 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 2.03 and compared to the observed operational leakage.

Reference 2 redefines the primary pressure boundary. The tube to tubesheet weld no longer functions as a portion of this boundary. The hydraulic expansion 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 steam generator operating conditions, including the most limiting accident conditions. The evaluation in Reference 2 determined that degradation in tubing below 11.2 inches from the top of the tubesheet does not require inspection or repair (plugging). The inspection of the portion of the tubes above 11.2 inches from the top of the tubesheet for tubes that have been hydraulically expanded in the tubesheet provides a high level of confidence that the structural and leakage performance criteria are maintained during normal operating and accident conditions.

WCAP-17071-P, 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. However, in response to a NRC staff request, NextEra commits to monitor for tube slippage as part of the steam generator tube inspection program.

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 that would invalidate assumptions in WCAP-17071-P. Therefore, NextEra commits to perform a one time verification of tube expansion locations to determine if any significant deviations exist from the top of tubesheet to the BET. If any significant deviations are found, the condition will be entered into the plant's corrective action program and dispositioned.

4.0 REGULATORY EVALUATION

4.1 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 2, provides reasonable assurance that the SG tubing remains capable of fulfilling its specific safety function of maintaining the reactor coolant pressure boundary. The NEI 97-06, Revision 2, SG performance criteria are:

- 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, cool down, and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design 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 loads.

- The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG, except for specific types of degradation at specific locations when implementing alternate repair criteria as documented in the Steam Generator Program technical specifications.

- The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day.

The proposed change defines the 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 SG operating conditions, including the most limiting accident conditions. The evaluation in Reference 2 determined that degradation in tubing below 11.2 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. NextEra has chosen to use an H* value of 13.1 inches for additional conservatism. As such, the Seabrook Station inspection program provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

4.2 Significant Hazards Consideration

NextEra 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. *The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.*

   The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed change that alters the steam generator (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 steam generator tube rupture

(SGTR) event, the steam line break (SLB), and the feed line break (FLB) postulated accidents.

During 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. Based on this design, the structural margins against burst, as discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," and Technical Specification 6.7.6.k, are maintained for both normal and postulated accident conditions.

The proposed change has no impact on the structural or leakage integrity of the portion of the tube outside of the tubesheet. The proposed change maintains structural and leakage integrity of the SG tubes consistent with the performance criteria of Technical Specification 6.7.6.k. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from tube degradation 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 severed tube. Therefore, the proposed change does not result in a significant increase in the consequences of a SGTR.

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

The leakage factor of 2.02 for Seabrook Station, for a postulated SLB/FLB, has been calculated as shown in Table 9-7 of Reference 2. However, NextEra will apply a factor of 2.03 to the normal operating leakage associated with the tubesheet expansion region in the condition monitoring (CM) and operational assessment (OA). The leakage factor of 2.03 is a bounding value for all SGs, both hot and cold legs, in Table 9-7 of Reference 2. 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. The assumed accident induced leak rate is 500 gallons per day (gpd) during a postulated steam line break in the faulted loop. Using the limiting leak rate factor of 2.03, this corresponds to an acceptable level of operational leakage of 246 gpd. Therefore, the technical specification leak rate limit of 150 gpd provides significant added margin against the 500 gpd accident analysis leak rate assumption.

No leakage factor will be applied to the locked rotor or control rod ejection transients due to their short duration.

For the CM assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.03 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the OA, the difference in the leakage between the allowable leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.03 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. *The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.*

The proposed change that alters the steam generator 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. *The proposed changes do not involve a significant reduction in the margin of safety.*

The proposed change limits the portion of the tube that must be inspected and repaired to the portion of the tube within the tubesheet necessary to maintain structural and leakage integrity under both normal and accident conditions. WCAP-17071-P identifies the specific inspection depth below which any type tube degradation shown to have no impact on the performance criteria in NEI 97-06 Rev. 2, "Steam Generator Program Guidelines."

The proposed change that alters the steam generator 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," and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," 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-17071-P defines 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 methodology for determining leakage as described in WCAP-17071-P shows that significant margin exists between an acceptable level of leakage during normal operating conditions (246 gpd) that ensures meeting the SLB accident-induced leakage assumption and the technical specification leakage limit of 150 gpd.

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

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

4.3 Conclusion

The hydraulically expanded portion of the tube 11.2 inches below the top of the tubesheet 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 SG operating conditions, including the most limiting accident conditions. WCAP-17071-P determined that degradation in this distance from the top of the tubesheet does not require plugging and serves as the basis for the limited tubesheet inspection criteria. NextEra has chosen to use an H* value of 13.1 inches for additional conservatism. WCAP-17071-P also shows that, upon implementation of the H* criterion, that the technical specification leakage limit of 150 gpd precludes unacceptable leakage during any postulated accident that models primary-to-secondary leakage.

In conclusion, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

NextEra 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, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed 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.

6.0 REFERENCES

1. NRC Letter "Seabrook Station, Unit 1 – Issuance of Amendment RE: Limited Inspection of the Steam Generator Tube Portion within the Tubesheet (TAC NO. MC85544)," September 29, 2006 (ML062630450).

2. Westinghouse Electric Company WCAP-17071-P," H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)" April 2009.

3. NEI 97-06, "Steam Generator Program Guidelines" Revision 2, May 2005.

4. EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7.

5. EPRI 1012987, Steam Generator Integrity Assessment Guidelines.

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

Attachment 2

Mark-up of the Technical Specifications (TS)

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 Amendment No. 120.

Listed below are the 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	Title	NextEra Energy Seabrook Letter	Date Submitted
LAR 09-01	License Amendment Request for Adoption of TSTF-511, Rev. 0, to Eliminate Working Hour Restrictions from Technical Specification 6.2.2 to Support Compliance with 10 CFR Part 26	SBK-L-09002	02/28/2009

The following TS pages are included in the attached markup:

Technical Specification	Title	Page
TS 3.7.6.k	Steam Generator (SG) Program	6-11 (info only) 6-12 (info only) 6-13 6-14
TS 6.8.1.7	Steam Generator Tube Inspection Report	6-21

INSERT 1

Tubes with service-induced flaws located greater than 13.1 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 13.1 inches below the top of the tubesheet shall be plugged upon detection.

INSERT 2

from 13.1 inches below the top of the tubesheet on the hot leg side to 13.1 inches below the top of the tubesheet on the cold leg side,

INSERT 3

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 13.1 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 2.03 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.

PROCEDURES AND PROGRAMS

This page contains no changes. Provided for information only.

6.7.6 (Continued)

j. Technical Specification (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 6.7.6j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

k. Steam Generator (SG) Program

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

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

This page contains no changes. Provided for information only

PROCEDURES AND PROGRAMS

6.7.6 (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 steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total or 500 gpd through any one SG.

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.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

shall

The following alternate tube repair criteria ~~may~~ be applied as an alternative to the 40% depth based criteria:

INSERT 1

~~1. During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, flaws identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet do not require plugging.~~

~~During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, all tubes with flaws identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the tubesheet shall be plugged.~~

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. ~~During refueling outage 11 and the subsequent operating cycles until the next scheduled inspection, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded from inspection when the alternate tube repair criteria in TS 6.7.6.k.c are implemented.~~ The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

INSERT 2

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

2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

INSERT 2

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

e. Provisions for monitoring operational primary to secondary leakage.

l. 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 Makeup Air and Filtration System (CREMAFS), 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.

c. Requirements for (i) determining the unfiltered air in-leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

6.8.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met. The CORE OPERATING LIMITS REPORT for each reload cycle, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

6.8.1.7 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 6.7.6.k, 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,

INSERT 3

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

SPECIAL REPORTS

6.8.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator within the time period specified for each report.

6.9 (THIS SPECIFICATION NUMBER IS NOT USED)

Attachment 3

Proposed TS Bases Change

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY (Continued)

rupture of a SG tube that relieves to the lower pressure secondary system. The analysis assumes that contaminated fluid is released to the atmosphere through the main steam safety valves or the atmospheric steam dump valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary-to secondary leakage from all SGs of 1 gallon per minute and 500 gallons per day from any one SG or is assumed to increase to these values as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.67 (Ref. 3), or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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, ~~between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet~~ ***from 13.1 inches below the top of the tubesheet on the hot leg side to 13.1 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 tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.7.6.k, "Steam Generator (SG) Program," 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.