Dominion Nuclear Connecticut, Inc.
5000 Dominion Boulevard, Glen Allen, Virginia 23060
Web Address: www.dom.com

Dominion®

PROPRIETARY INFORMATION – WITHHOLD UNDER 10CFR2.390

November 23, 2009

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

Serial No 09-525
NSSL/MLC R0
Docket No. 50-423
License No. NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION (TS) 6.8.4.g, "STEAM GENERATOR (SG) PROGRAM," AND TS 6.9.1.7, "STEAM GENERATOR TUBE INSPECTION REPORT" FOR ONE-TIME ALTERNATE REPAIR CRITERIA (H*)

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) hereby requests a one-time revision to Facility Operating License NPF-49 for Millstone Power Station Unit 3 (MPS3). This amendment request proposes to revise Technical Specification (TS) 6.8.4.g, "Steam Generator (SG) Program," to exclude a portion of the tubes below the top of the steam generator tubesheet from periodic steam generator tube inspections. Specifically, the change deletes the previous reference to the interim alternate repair criteria (IARC) applicable to Cycle 13 and adds information associated with a one-time alternate repair criteria (ARC) for Cycle 14. The amendment request also proposes to revise the reporting criteria in TS 6.9.1.7, "Steam Generator Tube Inspection Report," to remove reference to the previous Cycle 13 IARC and add reporting requirements specific to a Cycle 14, one-time ARC. These proposed changes are supported by Westinghouse Electric Company, LLC (Westinghouse) report WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," Revision 0, dated April 2009 (Enclosure 5).

This license amendment request (LAR) is based upon the application submitted by Southern Nuclear Operating Company, Inc. (SNC) for Vogtle Units 1 and 2, dated May 19, 2009. Vogtle functioned as the "lead plant" in preparing and submitting the application for the permanent alternate repair criteria (PARC) for steam generator tubes. Subsequent related correspondence was submitted by SNC on August 28, 2009 and September 11, 2009. The chronology associated with the Vogtle LAR is delineated in Attachment 1 to this letter.

This DNC submittal is based on maintaining structural and leakage integrity in the event of an accident. WCAP-17071-P recommends a 95% probability/50% confidence H* value of 11.2 inches; however, DNC has chosen to use an H* value of 13.1 inches for additional conservatism.

From a structural perspective, the 13.1 inch value of H* ensures that tube rupture or tube pull-out from the tubesheet will not occur in the event of an accident within the lifetime of the plant. Even in the event that all tubes in the steam generator have a 360

Enclosures 5, 6, 7, 8 and 9 contain information being withheld from public disclosure under 10CFR2.390. Upon separation, this page is decontrolled.

A001
NRR

degree severance at 13.1 inches, structural integrity of the steam generator tube bundle will be maintained.

From a leakage perspective, projections of accident-induced steam generator tube leakage are based on leakage rate factors applied to leakage detected during normal operation. The multiplication factor of 2.49 used for MPS3, bounds the expected increased leakage in the event of an accident.

As identified above, this LAR is modeled after the SNC submittal for Vogtle Units 1 and 2. This approach permits the use of previously reviewed correspondence which will facilitate the review of this submittal. Enclosure 1 to this DNC letter provides a discussion of the changes and is presented in three parts as follows:

- <u>PART A - Discussion of Change to Incorporate Permanent Alternate Repair Criteria</u>
  This section includes an MPS3-specific version of Attachment 1 to the Vogtle LAR dated May 19, 2009.

- <u>PART B – Response to NRC Request for Additional Information</u>
  This section includes an MPS3-specific version of Attachment 1 to SNC's August 28, 2009 RAI responses, noting the leakage factor increase from 2.03 to 2.49 that required revision of the proposed TS changes. The conclusions of the significant hazards consideration (SHC) were unchanged, however; the change from 2.03 to 2.49 (and associated reference change) was incorporated. The commitments affected by the RAI responses were also modified.

- <u>PART C - Discussion of Change from PARC to One-Time ARC</u>
  This section incorporates information from SNC's September 11, 2009 letter requesting the change from a PARC to a one-time ARC which resulted in a second revision to the proposed TS changes. Neither the SHC text nor the SHC conclusions were impacted. The commitments were affected by this change to reflect a one-time ARC.

The marked-up TSs in Enclosure 2 reflect the MP3-specific version of the final proposed TS changes from SNC's letter dated September 11, 2009 (ADAMS Ascension No. ML092540511). Enclosure 3 lists the MPS3 regulatory commitments consistent with the MP3 responses to the plant-specific RAI questions in Enclosure 1, Part B and the one-time ARC documented in Enclosure 1, Part C.

The proprietary information provided in Enclosures 5 through 7 is supported by affidavits signed by Westinghouse, the owner of the information. The affidavits set forth the basis on which the information may be withheld from public disclosure by the Commission and address with specificity, the considerations listed in paragraph (b)(4) of 10 CFR 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to Westinghouse, be withheld from public disclosure in accordance with 10 CFR 2.390 of the Commission's regulations. The affidavits are included in Enclosures 13 through 15 which also include proprietary information notices and copyright notices. Correspondence with respect to the copyright or proprietary aspects of the Westinghouse information noted above, or the supporting Westinghouse affidavits, should reference the applicable authorization letter 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. Redacted, non-proprietary versions of the Westinghouse supporting documentation are provided in Enclosures 10 through 12.

On other plant dockets (i.e., Vogtle Units 1 and 2), the NRC requested clarification regarding certain information marked as proprietary in WCAP-17071-P. Enclosure 8 is included in this transmittal to provide the clarification for MPS3. Enclosure 8 also provides proprietary and non-proprietary replacement pages for WCAP-17071-P and WCAP-17071-NP, respectively. The proprietary information contained in Enclosure 8 is supported by the affidavit contained in Enclosure 13; therefore, it is respectfully requested that this information be withheld from public disclosure in accordance with 10 CFR 2.390.

Enclosure 9 contains proprietary replacement pages for two figures that became illegible during the rendering and transmittal of Westinghouse RAI response dated August 12, 2009 (Enclosure 6). This proprietary information is supported by the affidavit contained in Enclosure 14; therefore, it is respectfully requested that this information be withheld from public disclosure in accordance with 10 CFR 2.390. These figures have been redacted in the non-proprietary version of this Westinghouse RAI response (Enclosure 11), therefore, non-proprietary replacement pages are not included.

DNC has evaluated the proposed permanent amendment request and determined that it does not involve an SHC as defined in 10 CFR 50.92. DNC concluded that this evaluation bounded the conditions associated with the proposed one-time amendment. The basis is included in Enclosure 4. DNC has also determined that operation with the proposed change will not result in any significant increase in the amount of effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed amendment is eligible for categorical exclusion from an environmental assessment as set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is needed in connection with the approval of the proposed change. The proposed change has been reviewed and approved by the Facility Safety Review Committee.

DNC requests NRC approval of the proposed license amendment by March 10, 2010 to support 3R13, which is currently scheduled to start on April 3, 2010. Once approved, the proposed changes will be implemented within 30 days of issuance of the amendment and prior to Mode 5 startup of MPS3.

DNC requests that the staff provide the specific questions remaining to be resolved and that the NRC continue the reviews needed to support approval of a permanent (versus one-time) ARC for the industry.

In accordance with 10 CFR 50.91(b), a copy of this LAR is being provided to the State of Connecticut.

Should you have any questions in regard to this submittal, please contact Ms. MaryLou Calderone at (860) 447-1791, extension 0681.

Sincerely,

J. Alan Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA
COUNTY OF HENRICO

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by J. Alan Price, who is Vice President – Nuclear Engineering of Dominion Nuclear Connecticut, Inc. He has affirmed before me that he is duly authorized to execute and file the foregoing document in behalf of that company, and that the statements in the document are true to the best of his knowledge and belief.

Acknowledged before me this 23rd day of November, 2009.

My Commission Expires: 4/30/13

Notary Public

GINGER L. ALLIGOOD
Notary Public
Commonwealth of Virginia
310847
My Commission Expires Apr 30, 2013

Commitments made in this letter:

1. DNC commits to monitor for tube slippage as part of the steam generator (SG) tube inspection program.

2. DNC commits to perform a one-time verification of the tube expansion to locate any significant deviations in the distance from the top of tubesheet to the bottom of the expansion transition (BET). If any significant deviations are found, the condition will be entered into the plant's corrective action program and dispositioned. Additionally, DNC commits to notify the NRC of significant deviations.

3. DNC commits to the following: For the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.49 and added to the total accident leakage from any other

source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region, will be divided by 2.49 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.

ATTACHMENT:

Chronology of Vogtle LAR

ENCLOSURES:

1. Discussion of Change
2. Marked-Up Technical Specifications Pages for One-Time Alternate Repair Criteria
3. List of Regulatory Commitments for One-Time Alternate Repair Criteria
4. Significant Hazards Consideration for One-Time Alternate Repair Criteria

Westinghouse Electric Company LLC Letters (Proprietary)

5. 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)," Revision 0, dated April 2009. (See Enclosure 12 for affidavit)
6. Westinghouse Electric Company LLC, LTR-SGMP-09-100 P-Attachment, Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators, dated August 12, 2009. (See Enclosure 13 for affidavit)
7. Westinghouse Electric Company LLC, LTR-SGMP-09-109 P-Attachment, Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators, dated August 25, 2009. (See Enclosure 14 for affidavit)
8. Westinghouse Electric Company LLC, LTR-RCPL-09-131, WCAP-17071-P, Rev. 0 Proprietary Information Clarification, dated August 27, 2009. (See Enclosure 12 for affidavit) *Also includes non-proprietary replacement pages for WCAP-17071-NP.*
9. Westinghouse Electric Company LLC, LTR-SGMP-09-121, Replacements for Illegible Pages in Prior RAI Response (Reference 1), dated August 27, 2009. (See Enclosure 13 for affidavit)

Westinghouse Electric Company LLC Letters (Non-Proprietary)

10. Westinghouse Electric Company LLC, WCAP-17071-NP, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," Revision 0, dated April 2009. (Non-Proprietary)

11. Westinghouse Electric Company LLC, LTR-SGMP-09-100 NP-Attachment, Response to NRC Request for Additional Information on H*; Model F and Model D5 Steam Generators, dated August 12, 2009. (Non-Proprietary)

12. Westinghouse Electric Company LLC, LTR-SGMP-09-109 NP-Attachment, Response to NRC Request for Additional Information on H*; RAI #4; Model F and Model D5 Steam Generators, dated August 25, 2009. (Non-Proprietary)

Westinghouse Electric Company LLC Authorization Letters:

13. Westinghouse Electric Company LLC, CAW-09-2569, "Application for Withholding Proprietary Information from Public Disclosure," dated May 4, 2009.

14. Westinghouse Electric Company LLC, CAW-09-2632, "Application for Withholding Proprietary Information from Public Disclosure," dated August 13, 2009.

15. Westinghouse Electric Company LLC, CAW-09-2659, "Application for Withholding Proprietary Information from Public Disclosure," dated August 27, 2009.

cc: U.S. Nuclear Regulatory Commission
Region I Regional Administrator
475 Allendale Road
King of Prussia, PA 19406-1415

C. J. Sanders
NRC Project Manager
U.S. Nuclear Regulatory Commission, Mail Stop 08B3
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Millstone Power Station

Director
Bureau of Air Management
Monitoring and Radiation Division
Department of Environmental Protection
79 Elm Street
Hartford, CT 06106-5127

ATTACHMENT 1

CHRONOLOGY OF VOGTLE LAR

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

CHRONOLOGY OF VOGTLE LAR

SNC LAR Dated May 19, 2009

On May 19, 2009, SNC submitted a license amendment request for Vogtle Units 1 and 2 (Reference 1) to permanently revise TS 5.5.9, "Steam Generator (SG) Program," to exclude portions of the tube below the top of the steam generator tubesheet from periodic steam generator tube inspections. This permanent change was supported by WCAP-17071-P. The submittal included WCAP-17071-P, TS mark-ups, and the significant hazards consideration (SHC). The SHC concluded that the proposed change did not involve a significant increase in the probability or consequences of an accident previously evaluated, did not create the possibility of a new or different kind of accident from any accident previously evaluated, and did not involve a significant reduction in any margin of safety.

SNC RAI Response Dated August 28, 2009

As part of the review of SNC's LAR dated May 19, 2009, the NRC issued two requests for additional information (RAIs) to SNC on July 10, 2009 (Reference 2) and August 5, 2009 (Reference 3). The July 10, 2009 RAI contained twenty-four (24) questions, with three of the questions being site-specific. The August 5, 2009 RAI contained three questions related to Questions 4, 20 and 24 from the July 10, 2009 RAI, as well as one additional site-specific question (NRC Question 25). With the exception of the four site-specific questions, Westinghouse developed responses to the questions in LTR-SGMP-09-100-P (Enclosure 6), with the response to RAI #4 being provided under a separate cover (LTR-SGMP-09-109-P - Enclosure 7). RAIs submitted to Wolf Creek (Reference 4), Byron/Braidwood, Comanche Peak and Seabrook, also included one additional question (NRC Question 26) that was not included on the Vogtle docket. This question was also addressed by Westinghouse in LTR-SGMP-09-100-P (Enclosure 6). SNC responses to the four site-specific RAI questions, as well as the Westinghouse RAI responses, were submitted to the NRC in two separate letters (Serial Nos. NL-09-1265 and NL-09-1375), both dated August 28, 2009 (References 5 and 6).

As described in the response to RAI Question 24, a change was made to increase the leak rate factor from 2.03 to 2.48. The leak rate factor is applied to the operational leak rate to determine the accident leakage due to tube flaws contained within the tubesheet. The basis for the leak rate factor change was to ensure the accident leakage from a feedwater line break (FLB) accident, when it is assumed to be a heat-up event, remains bounded by the site accident induced leakage limit of 1.0 gallon per minute (gpm) at room temperature (gpmRT). The increased leak rate factor resulted in changes to the proposed reporting requirements in TS 5.6.10. Consequently, the Vogtle RAI response dated August 28, 2009 (Reference 5) included revised markups to TS 5.6.10.

The accident induced leak rate limit is 1.0 gpm. The TS operational leak rate is 150 gallons per day (gpd) (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The steam line break (SLB)/FLB leak rate ratio is only 2.48, resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

The revised leak rate factor did not affect the structural H* analysis because the H* structural analysis is bounded by normal operating conditions and not by accident conditions. The leak rate factor was not used in the structural H* analysis and there was no change to the normal operating conditions as previously evaluated; therefore, the H* length remains unchanged.

Based on the above information and a review of the enclosures, the additional information provided did not expand the scope of the application as originally published and did not impact the conclusions of the NRC staff's original proposed SHC determination, as noticed in the Federal Register (74 FR 40240) on August 11, 2009. Therefore, no changes to the conclusions of the SHC were required.

<u>SNC Letter Dated September 11, 2009</u>
On September 2, 2009, in a teleconference between NRC staff and industry personnel, NRC staff indicated their concerns with eccentricity of the tubesheet tube bore in normal and accident conditions (RAI Question 4 of the July 10, 2009 letter and RAI Question 1 of the August 5, 2009 letter), have not been completely resolved. The staff further indicated that there was insufficient time to resolve these issues to support approval of the PARC amendment request for the fall 2009 refueling outages. The staff concluded, however, that the information submitted to date was sufficient to support a one-time TS amendment to provide sufficient time to resolve the eccentricity concerns. As such, SNC requested that the TS changes proposed in the August 28, 2009 RAI response (Reference 5) be revised to be a one-time change to support the upcoming refueling outage and subsequent operating cycle.

REFERENCES

1) Letter dated May 19, 2009, M. J. Ajluni, Southern Nuclear Operating Company, Inc., to USNRC, "Vogtle Electric Generating Plant, License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria."

2) Letter dated July 10, 2009, from D. Wright, USNRC, to M. J. Ajluni, Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2, Request for Additional Information Regarding Steam Generator Program (TAC NOS. ME1339 and ME1340)."

3) Letter dated August 5, 2009, D. Wright, USNRC, to M. J. Ajluni, Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2, Request for Additional Information Regarding Steam Generator Program (TAC NOS. ME1339 and ME1340)."

4) Letter dated August 11, 2009, B. K. Singal, USNRC, to R. A. Muench, Wolf Creek Nuclear Operating Corporation, "Wolf Creek Generating Station – Request for Additional Information Regarding the Permanent Alternate Repair Criteria License Amendment Request (TAC No. ME1393)."

5) Letter NL-09-1265 dated August 28, 2009, M. J. Ajluni, Southern Nuclear Operating Company, Inc., to USNRC, "Vogtle Electric Generating Plant, Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria.

6) Letter NL-09-1375 dated August 28, 2009, M. J. Ajluni, Southern Nuclear Operating Company, Inc., to USNRC, "Vogtle Electric Generating Plant, Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria."

ENCLOSURE 1

DISCUSSION OF CHANGE

INTRODUCTION

PART A: Discussion of Change to Incorporate Permanent Alternate Repair Criteria

PART B: Response to Request for Additional Information

PART C: Discussion of Change from PARC to One-Time ARC

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

INTRODUCTION

This Dominion Nuclear Connecticut, Inc. (DNC) license amendment request (LAR) is modeled after an industry initiative that employed Vogtle as the "lead plant." The licensing process employed by Vogtle can be divided into three separate activities, with associated submittals to the Nuclear Regulatory Commission (NRC). This DNC submittal incorporates the information submitted by Vogtle (via multiple submittals over a period of several months).

This enclosure captures each of the Vogtle submittals in sequence in a single document in order to follow the path established by the "lead plant." The enclosure is divided into three parts as follows:

Part A addresses the original LAR based on Westinghouse Electric Company, LLC (Westinghouse) report WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," Revision 0, dated April 2009 (Enclosure 5). Part A, therefore, documents the original request for permanent alternate repair criteria (PARC).

Part B addresses all of the responses to NRC's requests for additional information (RAIs) and the resultant impact on the original document in Part A.

Part C addresses the subsequent changes associated with the shift from a PARC to a one-time alternate repair criteria (ARC).

Since evolution of the LAR from a PARC to a one-time ARC resulted in changes to the marked-up technical specifications (TSs), regulatory commitments and the basis for the significant hazards consideration (SHC), the final versions of each of these documents are included as Enclosures 2, 3 and 4, respectively.

PART A

DISCUSSION OF CHANGE TO INCORPORATE PERMANENT ALTERNATE REPAIR CRITERIA

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, DNC hereby requests a permanent revision to Facility Operating License NPF-49 for Millstone Power Station Unit 3 (MPS3). This LAR proposes to revise TS 6.8.4.g, "Steam Generator (SG) Program," to exclude a portion of the tubes below the top of the steam generator tubesheet from periodic steam generator tube inspections. The change deletes information associated with the interim alternate repair criteria (IARC) for Cycle 13 and adds information associated with the PARC. The amendment request also proposes to revise the reporting criteria in TS 6.9.1.7, "Steam Generator Tube Inspection Report," to remove reference to the Cycle 13 IARC and add reporting requirements specific to the PARC. These proposed changes are supported by WCAP-17071-P.

This H* submittal is based on maintaining structural and leakage integrity in the event of an accident. WCAP-17071-P recommends a 95% probability/50% confidence H* value of 11.2 inches; however, DNC has chosen to use an H* value of 13.1 inches for additional conservatism.

From a structural perspective, the 13.1 inch value of H* ensures tube rupture or tube pull-out from the tubesheet will not occur in the event of an accident within the lifetime of the plant. Even in the event that all tubes in the steam generator have a 360 degree severance at 13.1 inches, structural integrity of the steam generator tube bundle will be maintained. This assumption bounds the current status of the MPS3 steam generators with sufficient margin.

The NRC previously issued License Amendment (LA) 245 for MPS3 (Part A, Reference 1) which approved the IARC that requires full-length inspection of the tubes within the tubesheet but does not require plugging tubes if any circumferential cracking observed in the region greater than 17 inches from the top of the tubesheet is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads. LA 245 was approved to support Refueling Outage 12 (3R12) and expires at the end of the current operating cycle (Cycle 13).

2.0 DETAILED DESCRIPTION

Proposed Changes to TS 6.8.4.g.c: Deleted text is struck through and added text is italicized and bold.

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 MPS3 Refueling Outage 12 and the subsequent operating cycle, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.~~

   ~~Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.~~

   ~~When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.~~

   ~~When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.~~

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.

Proposed Changes to TS 6.8.4.g.d: Deleted text is struck through and added text is italicized and bold.

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,*** ~~the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet,~~ and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

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

3. If crack indications are found in any SG tube ***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 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.

Proposed Changes to TS 6.9.1.7: Deleted text is struck through and added text is italicized and bold.

6.9.1.7 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with TS 6.8.4.g, 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. Following completion of an inspection performed in Refueling Outage 12 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet, and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 6.8.4.g.c,~~

~~j. Following completion of an inspection performed in Refueling Outage 12 (and any inspections performed in the subsequent operating cycle), the primary-to-secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG, the entire primary-to-secondary LEAKAGE should be conservatively assumed to be from one~~

~~steam generator) during the cycle preceding the inspection which is the subject of the report, and~~

~~k. Following completion of an inspection performed in Refueling Outage 12 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below 17 inches from the top of the tubesheet for the most limiting accident in the most limiting steam generator.~~

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; and***

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 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.***

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

MPS3 is a four loop Westinghouse designed plant with Model F steam generators having 5626 tubes in each steam generator. A total of 159 tubes are currently plugged in all four steam generators. The design of the steam generators include 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 6.8.4.g, "Steam Generator (SG) Program," Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," (Part A, Reference 2); EPRI "Pressurized Water Reactor Steam Generator Examination Guidelines," (Part A, Reference 3); EPRI "Steam Generator Integrity Assessment Guidelines," (Part A, Reference 4); ER-AP-SGP-101, "Steam Generator Program," (Part A, Reference 8), and the results of the degradation assessments required by the Steam Generator 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 MPS3. 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 steam generator tubes during their fall 2004 outage. NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," (Part A, Reference 5), 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.

DNC policies and programs require the use of applicable industry operating experience in the operation and maintenance of MPS3. 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" – Part A, Reference 7). Since the MPS3 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 MPS3 to identify tube indications similar to those reported at Catawba within the hot leg tubesheet region if similar inspections are performed during the spring 2010 refueling outage.

Potential inspection plans for the tubes and tube welds underwent intensive industry discussions in March 2005. The findings in the Catawba steam generator tubes present two distinct issues with regard to the steam generator tubes at MPS3:

1) Indications in internal bulges and overexpansions within the hot leg tubesheet; and

2) Indications at the elevation of the tack expansion transition

Prior to each steam generator 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 MPS3 steam generators. 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, MPS3 revised the steam generator inspection plan for Refueling Outage 10 (fall 2005) and subsequent refueling outages to include sampling of bulges and overexpansions within the tubesheet region on the hot leg side. The sample was based on the guidance contained in the EPRI "Pressurized Water Reactor Steam Generator Examination Guidelines" and TS 6.8.4.g, "Steam Generator (SG) Program." Degradation was not detected in the tubesheet region in Refueling Outage 10 or Refueling Outage 11.

For Refueling Outage 12 (fall 2008), an IARC was approved which revised TS 6.8.4.g. The IARC required an inspection to the full depth of the tubesheet and allowed axial cracks and circumferential cracks, less than a specified extent, to remain in service at the tube ends. Indications were identified in the hot leg tube ends of steam generators 'A' and 'C', which required tube end inspection scope expansions that included steam generators 'B' and 'D'. Indications were observed at the hot leg tube ends in all four steam generators and in one cold leg tube end in steam generator 'D'. All indications were within approximately 0.75 inches from the tube end. Indications with circumferential extent greater than 94 degrees and mixed-mode indications were plugged. All axial and circumferential oriented indications 94 degrees or less in circumferential extent, were left in service consistent with the criteria provided in the IARC. Axial indications and indications with circumferential extent of up to, and including 94 degrees, do not challenge the structural and leakage integrity requirements of NEI 97-06.

As a result of these potential issues, and to prevent the unnecessary plugging of additional tubes in the MPS3 steam generators, DNC is proposing a change to TS 6.8.4.g to limit the steam generator tube inspection and repair (plugging) to the portion of 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. In addition, this LAR proposes to revise TS 6.9.1.7, "Steam Generator Tube Inspection Report," to provide reporting requirements specific to this PARC.

3.2 Evaluation

To preclude unnecessarily plugging tubes in the MPS3 steam generators, an evaluation was performed to identify the safety significant portion of the tube within the tubesheet necessary to maintain structural and leakage integrity in both normal and accident conditions. Tube inspections will be limited to identifying and plugging degradation in the safety significant portion of the tubes. The technical evaluation for the inspection and repair methodology is provided in WCAP-17071-P (Enclosure 5). 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 MPS3 Model F steam generator, 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. 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 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 WCAP-17071-P. The tube-to-tubesheet radial contact pressure provides resistance to tube pullout and resistance to leakage during plant operation and transients.

Primary-to-secondary leakage from tube degradation in the tubesheet area is accounted for 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 TSs limit primary-to-secondary leakage to 150 gallons per day (gpd) through any one steam generator. In addition, leakage is limited by a leakage factor of 2.03 which is the bounding value for all steam generators, both hot and cold legs, in Table 9-7 of Enclosure 5. For MPS3, 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 steam generator 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 from 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," (Part A, Reference 6) 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 accident induced leak rate limit is 1.0 gallon per minute (gpm). The TS operational leak rate limit is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB leak rate ratio is only 2.03 resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

Plant-specific operating conditions are used to generate the overall leakage factor ratios that are to be 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 steam generator, 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 cooldown 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 viscosity 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 2.03, which 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 design specification transients which result in increased temperatures in the steam generator 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 an SLB and FLB, the length of time that a plant with Model F steam generators 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 gpm, the leak rate for a locked rotor event can be integrated over a minute to compare to the limit. Time integration permits an increase in acceptable leakage during the time of peak pressure differential by approximately a factor of two because of the short duration (less than 30 seconds) of the elevated pressure differential. This translates into an effective reduction in leakage factor by the same factor of two for the locked rotor event. Therefore, for the locked rotor event, the leakage factor of 1.78 (Table 9-7, Enclosure 5) for MPS3 is adjusted downward to a factor of 0.89. Similarly, for the control rod ejection event, the duration of the elevated pressure differential is less than 10 seconds. Thus, the peak leakage factor may be reduced by a factor of six from 2.69 to 0.45.

The plant transient response following a full power double-ended main feedwater line rupture corresponding to "best estimate" initial conditions and operating characteristics indicates that the transient for a Model F steam generator exhibits a cooldown characteristic instead of a heat-up transient as generally presented in steam generator design transients and in the UFSAR Chapter 15.0 safety analysis. 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 steam generator, the FLB transient curve (Figure 9-5, Enclosure 5) 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 requirements and combines 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 feature actuation would be less severe. This would result in much earlier reactor trip and greatly increase the steam generator 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 control system (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 2.03 for a postulated SLB/FLB has been calculated as shown in Table 9-7 of Enclosure 5. The leakage factor of 2.03 is a bounding value for all steam generators, both hot and cold legs. Specifically, for the condition monitoring (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 operational assessment (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.

WCAP-17071-P redefines the primary pressure boundary. The tube-to-tubesheet weld no longer functions as a portion of this boundary. The hydraulically expanded portion of the tube into the tubesheet over the H* distance now functions as the primary pressure boundary in the area of the tube and tubesheet, maintaining the structural and leakage integrity over the full range of steam generator operating conditions, including the most limiting accident conditions. The evaluation in WCAP-17071-P 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. Notwithstanding the conclusions of WCAP-17071-P, DNC has chosen to use an H* value of 13.1 inches for additional conservatism.

WCAP-17071-P, Section 9.8, provides a review of leak rate susceptibility due to tube slippage and concluded that the tubes are fully restrained against motion under very conservative design and analysis assumptions such that tube slippage is not a credible event for any tube in the bundle. However, in response to a NRC staff request, DNC 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 the tubesheet that would invalidate assumptions in WCAP-17071-P. Therefore, DNC 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 steam generator 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, established reactor siting criteria, with respect to the risk of public exposure to the release of radioactive fission products. Accidents involving leakage or tube burst of steam generator 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 steam generators as risk significant components because they are relied upon to remain functional during and after design basis events. Steam generators 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 steam generator tubing remains capable of fulfilling its specific safety function of maintaining the RCPB. The steam generator performance criteria from NEI 97-06, Revision 2 and MPS3 TS 6.8.4.g, "Steam Generator (SG) Program," are:

- 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, 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 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 steam generator tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all steam generators and leakage rate for an individual steam generator. Leakage is not to exceed 1.0 gpm per steam generator, 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 steam generator shall be limited to 150 gpd.

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 WCAP-17071-P determined that degradation in tubing below the safety significant portion of the tube does not require plugging and serves as the bases for the tubesheet inspection program. As such, the MPS3 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

This LAR proposes a change to TS 6.8.4.g, "Steam Generator (SG) Program," to exclude portions of the tubes within the tubesheet from periodic steam generator inspections. In addition, this LAR proposes a change to TS 6.9.1.7, "Steam Generator Tube Inspection Report" to remove reference to the previous (Cycle 13) IARC and provide revised reporting requirements. 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 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 Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," (Part A, Reference 2) performance criteria.

DNC 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 steam generator inspection criteria and the steam generator inspection 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 steam generator tube inspection and repair criteria are the steam generator tube rupture (SGTR) event and the feedline break (FLB) postulated accidents.

During the SGTR event, the required structural integrity margins of the steam generator tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the tube-to-tubesheet joint. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, as discussed in RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes,"(Part A, Reference 6) 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 integrity of the steam generator tubes and does not affect other systems, structures, components, or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. However, primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed changes since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter. Therefore, the proposed changes do not result in a significant increase in the consequences of a SGTR.

The consequences of a steam line break (SLB) are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the steam generator creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., a SLB) is limited by flow restrictions. These restrictions result from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications.

The leakage factor of 2.03 for MPS3, for a postulated SLB/FLB, has been calculated as shown in Table 9-7 of Enclosure 5. The leakage factor of 2.03 is a bounding value for all steam generators, both hot and cold legs, in Table 9-7. Specifically, for the condition monitoring (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 operational assessment (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.

The probability of a SLB is unaffected by the potential failure of a steam generator tube as the failure of the tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial and or circumferentially orientated cracks occurring 13.1 inches below the top of the tubesheet. The accident induced leak rate limit is 1.0 gpm. The TS operational leak rate is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB leak rate ratio is only 2.03 resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

Therefore, 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 steam generator inspection criteria and the steam generator inspection 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 accident previously evaluated.

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

Response: No

The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria maintains the required structural margins of the steam generator tubes for both normal and accident conditions NEI 97-06, Revision 2, "Steam Generator Program Guidelines" and RG 1.121, are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting 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, WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," 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 provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

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

4.3 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. WCAP-17071-P 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 a 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 CONSIDERATIONS

DNC has evaluated the proposed amendment for environmental considerations. The review has resulted in the determination 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 forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1) NRC letter "Millstone Power Station, Unit No. 3 – Issuance of Amendment Regarding Changes to Technical Specification (TS) Section 6.8.4.g, "Steam Generator Program" and Section 6.9.1.7, "Steam Generator Tube Inspection Report" (TAC No. MD8736)," dated September 30, 2008.

2) NEI 97-06, Rev. 2, "Steam Generator Program Guidelines," May 2005.

3) EPRI "Pressurized Water Reactor Steam Generator Examination Guidelines."

4) EPRI "Steam Generator Integrity Assessment Guidelines."

5) NRC Information Notice (IN) 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds."

6) Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976, (ADAMS Accession No. ML003739366).

7) NRC Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections," dated August 30, 2004.

8) ER-AP-SGP-101, "Steam Generator Program," Revision 3.

PART B

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

1.0 SUMMARY DESCRIPTION

As previously discussed, Vogtle functioned as the "lead plant" for the licensing of a permanent alternate repair criteria (PARC) for steam generator tubes. As part of the review of SNC's submittal, the NRC issued two requests for additional information (RAIs) to SNC on July 10, 2009 (Part B, Reference 2) and August 5, 2009 (Part B, Reference 3). The July 10, 2009 RAI contained twenty-four (24) questions, with three of the questions being site-specific. The August 5, 2009 RAI contained three questions related to Questions 4, 20 and 24 from the July 10, 2009 RAI, as well as one additional site-specific question (NRC Question 25). With the exception of the four site-specific questions, Westinghouse developed responses to the questions in LTR-SGMP-09-100-P (Enclosure 6), with the response to RAI #4 being provided under a separate cover (LTR-SGMP-09-109-P - Enclosure 7). RAIs submitted to Wolf Creek (Part B, Reference 4), Byron/Braidwood, Comanche Peak and Seabrook, also included one additional question (NRC Question 26) that was not included on the Vogtle docket. This question was also addressed by Westinghouse in LTR-SGMP-09-100-P (Enclosure 6). SNC responses to the four site-specific RAI questions, as well as the Westinghouse RAI responses, were submitted to the NRC in two separate letters, both dated August 28, 2009 (Part B, References 5 and 6).

2.0 DNC RESPONSES TO NRC RAIs

Provided below are the DNC responses to the twenty-six (26) RAI questions from the NRC. The NRC question numbers below refer to the NRC letter dated July 10, 2009 (Part B, Reference 2) unless otherwise noted.

<u>NRC Question 1</u>
Reference 1, page 6-21, Table 6-6. This table contains a number of undefined parameters and some apparent inconsistencies with Table 5-2 on page 5-6. Please define the input parameters in Table 6-6.

<u>DNC Response</u>
The Question 1 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

<u>NRC Question 2</u>
Reference 1, page 6-23, Section 6.2.2.2. Why was the finite element analysis not run directly with the modified temperature distribution rather than running with the linear distribution and scaling the results?

<u>DNC Response</u>
The Question 2 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 3

Reference 1, page 6-38, Section 6.2.3. Why is radial displacement the "figure of merit" for determining the bounding segment? Does circumferential displacement not enter into this? Why is the change in the tube hole diameter not the "figure of merit?"

DNC Response

The Question 3 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 4

Reference 1, page 6-69. In Section 6.2.5.3, it is concluded that the tube outside diameter and the tubesheet tube bore inside diameter always maintain contact in the predicted range of tubesheet displacements. However, for tubes with through wall cracks at the H distance, there may be little or no net pressure acting on the tube for some distance above H*. In Tables 6-18 and 6-19, the fourth increment in the step that occurs two steps prior to the last step suggests that there may be no contact between the tube and tubesheet, over a portion of the circumference, for a distance above H*. Is the conclusion in Section 6.2.5.3 valid for the entire H* distance, given the possibility that the tubes may contain through wall cracks at that location?*

In August 5, 2009 letter, the following additional information was requested as part of the response to RAI 4:

a. Clarify the nature of the finite element model ("slice" model versus axisymmetric SG assembly model) used to generate the specific information in Tables 6-1,2, and 3 (and accompanying graph entitled "Elliptical Hole Factors") of Reference 6-15. What loads were applied? How was the eccentricity produced in the model? (By modeling the eccentricity as part of the geometry? By applying an axisymmetric pressure to the inside of the bore?) Explain why this model is not scalable to lower temperatures.

b. Provide a table showing the maximum eccentricities (maximum diameter minus minimum diameter) from the 3 dimensional (3-D) finite element analysis for normal operating and steam line break (SLB), for model F and model D5 SGs.

c. In Figure 2 of the White Paper, add a plot for the original relationship between reductions in contact pressure and eccentricity as given in Reference 6-15 in the graph accompanying Table 6-3. Explain why this original relationship remains conservative in light of the new relationship. Explain the reasons for the differences between the curves.

d. When establishing whether contact pressure increases when going from normal operating to SLB conditions, how can a valid and conservative comparison be made if the normal operating case is based on the original delta contact pressure versus eccentricity curve and the SLB case is based on the new curve?

DNC Response

The Question 4 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question (as originally stated in Reference 2 and amplified in Reference 3).

NRC Question 5

Reference 1, page 6-87 - Are the previously calculated scale factors and delta D factors in Section 6.3 conservative for steam line break and feed line break? Are they conservative for an intact divider plate assumption? Are they conservative for all values of primary pressure minus crevice pressure that may exist along the H distance for intact tubes and tubes with through wall cracks at the H* distance? How is tube temperature (T_T) on page 6-87 determined? For normal operating conditions, how is the T_T assumed to vary as function of elevation?*

DNC Response

The Question 5 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 6

Reference 1, page 6-97, Figure 6-75 - Contact pressures for nuclear plants with Model F SGs are plotted in Figure 6-75, but it is not clear what operating conditions are represented in the plotted data, please clarify.

DNC Response

The Vogtle Question 6 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 7

Reference 1, page 6-113, Reference 6-5 - This reference seems to be incomplete; please provide a complete reference.

DNC Response

The Question 7 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 8

Reference 1, page 6-113, Reference 6-15 - Table 6-3 in Reference 6-15 (SM-94-58, Rev 1) appears inconsistent with Table 6-2 in the same reference. Explain how the analysis progresses from Table 6-2 to Table 6-3.

DNC Response

The Question 8 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 9

Reference 1, page 8-9, Figure 8-1 - There is an apparent discontinuity in the plotted data of the adjustment to H for distributed crevice pressure, please provide any insight you may have as to why this apparent discontinuity exists.*

DNC Response

The Question 9 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 10

Reference 1, page 8-6, Section 8.1.4 - Clarify whether the "biased" H distributions for each of the four input variables are sampled from both sides of the mean H* value during the Monte Carlo process, or only on the side of the mean H* value yielding an increased value of H*.*

DNC Response

The Question 10 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question. Legible copies of Figures RAI 10-1 and 10-2 are included in Enclosure 9.

NRC Question 11

Reference 1, page 8-14, Figure 8-6 - The legend for one of the interactions shown between α_{TS} and E_{Ts} appears to have a typo in it, please review and verify that all values shown in the legend are correct.

DNC Response

The Question 11 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 12

Reference 1, page 8-20, Case S-4 - Why does the assumption of a 2-sigma value for the coefficient of thermal expansion of the tube (αT) and the tubesheet (αTS) to determine a "very conservative biased mean value of H" conservatively bound the interaction effects between αT and αTS? Describe the specifics of how the "very conservative biased mean value of H*," as shown in Table 8-4, was determined.*

DNC Response

The Question 12 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 13

Reference 1, page 8-22, Case M-5 - The description for this case seems to correspond to a single tube H estimate rather than a whole bundle H* estimate. How is the analysis performed for a whole bundle H* estimate?*

DNC Response

The Question 13 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 14

Reference 1, page 8-22, Case M-5 states, "Interaction effects are included because the 4.285 sigma variations were used that already include the effective interactions among the variables." Case M-5 also states that the 4.285 sigma variations come from Table 8-2. However, Table 8-2 does not appear to include interactions among the variables. Explain how the 4.285 sigma variations include the effect of interactions among the variables.

DNC Response
The Question 14 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 15
Reference 1, page 8-22, Case M-6, first bullet - Should the words "divided by 4.285" appear at the end of the sentence?

DNC Response
The Question 15 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 16
Reference 1, page 8-23, Case M-7 - Was the "2 sigma variation of all variables" divided by a factor of 2?

DNC Response
The Question 16 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 17
Reference 1, page 8-23, Case M-7 - Explain how this case includes the interaction effects between the two principle variables, αT and αTS.

DNC Response
The Question 17 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 18
Reference 1, page 8-25, Table 8-4 - Explain why the mean H calculated in the fifth case does not require the same adjustments, as noted by the footnotes, that all other cases in the table require.*

DNC Response
The Question 18 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 19
Reference 1, page 8-25, Table 8-4 - Verify the mean H shown in the last case in the table.*

DNC Response
The Question 19 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

NRC Question 20

Section 8 of Reference 1 - The variability of H with all relevant parameters is shown in Figure 8-3. The interaction between αT and αTS are shown in Figure 8-5. Please explain why the direct relationships shown in these two figures were not sampled directly in the Monte Carlo analysis, instead of the sampling method that was chosen. Also, please explain why the sampling method chosen led to a more conservative analysis than directly sampling the relationships in Figures 8-3 and 8-5.*

In August 5, 2009 letter, the following additional information was requested as part of response to RAI 20, include discussion of main steam line break (SLB) and whether it continues to be less limiting, from maximum H perspective, than three times normal operating pressure.*

DNC Response

The Question 20 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question. The response to the August 5, 2009 question (Part B, Reference 2, Question 2) is contained within the body of the response to Question 20.

NRC Question 21 (site-specific)

The limiting leakage factor for VEGP is greater than 2.0 per Reference 1. The reporting requirement proposed by VEGP only requires them to report if they use a leakage factor of less than 2.0. The NRC staff understands that the licensee does not want to give a false impression that it can measure very small leak rates; however, the NRC staff feels it is appropriate for the licensee to use a number that bounds the plant-specific limiting leakage factor in Reference 1. Please discuss your plans to incorporate a limiting leakage factor that bounds the value in Reference 1.

DNC Response

As described in the response to Question 24 in Westinghouse LTR-SGMP-09-100-P (Enclosure 6), a change is being made to increase the leak rate factor for MPS3 from 2.03 to 2.49 (Reference Table RAI 24-2 from Enclosure 6). (See also Enclosure 3, Commitment 3)

NRC Question 22 (site-specific)

In the May 19, 2009, letter, VEGP commits to monitor for tube slippage as part of the SG tube inspection program. The "due date/event" is prior to the start of Refueling Outage 1R15. It is not clear whether the planned monitoring will be performed once and whether it only applies to Unit 1. The commitment should be modified to indicate that the tube slippage will be monitored at both units during every SG tube inspection outage.

DNC Response

Tube slippage monitoring is applicable to MPS3. DNC commits to monitor for tube slippage as part of the MPS3 steam generator tube inspection program. (See Enclosure 3, Commitment 1)

NRC Question 23 (site-specific)

In the May 19, 2009, letter, VEGP commits to determine the position of the bottom of the expansion transition in relation to the top of the tubesheet and to enter "any significant *deviation" into their corrective action program. This is a one-time verification prior to implementation of H*. The commitment should be modified to also include a commitment to notify the NRC staff if significant deviations in the location of the bottom of the expansion transition relative to the top of the tubesheet are detected.*

DNC Response
DNC commits to perform a one-time verification of the tube expansion to locate any significant deviations in the distance from the top of the tubesheet to the bottom of the expansion transition (BET). If any significant deviations are found, the condition will be entered into the plant's corrective action program and dispositioned. Additionally, DNC commits to notify the NRC of significant deviations. (See Enclosure 3, Commitment 2)

NRC Question 24
Reference 1, Page 9-6, Section 9.2.3.1 - The feedwater line break heat-up transient is part of the plant design and licensing basis. Thus, it is the NRC staffs position that H and the "leakage factors," as discussed in Section 9.4, should include consideration of this transient. Explain why the proposed H* and leakage factor values are conservative, even with consideration of the feedwater line break heat-up transient.*

In August 5, 2009 letter, the following additional information was requested as part of the response to RAI 24, VEGP should address, specifically, the feed line break (FLB) heatup transient in the final safety analysis report as it is part of the licensing bases and needs to be addressed. Please provide a rationale to justify basing the leakage factor on SLB, or commit to a leakage factor based on the FLB heatup transient.

DNC Response
The Question 24 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question. The response to the August 5, 2009 question (Part B, Reference 3, Question 3), is contained within the body of the response to Question 24.

NRC Question 25 (Question 4 of August 5, 2009 Letter – Part B, Reference 3)
During review of the VEGP amendment request, it was noticed that the regulatory commitment regarding use of the leakage factor had been stated in the body of the document (page E1-11) but had been omitted from the list of regulatory commitments in Enclosure 4. Since the final leakage factor may change based on the FLB analysis (question 4 above), the proper factor will need to be used in the regulatory commitment. Please include the commitment, with the appropriate leakage factor, in the list of regulatory commitments in Enclosure 4 of the application.

> *For the condition monitoring (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 operational assessment (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. An administrative limit will be established to not exceed the calculated value.*

DNC Response
DNC has revised Commitment 3 in Enclosure 3 to read as follows: "For the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.49 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.49 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value."

NRC Question 26 (Question 6 from Part B, Reference 4)
Reference 1, page 6-87: Please provide information on how the tube temperature (T_T) on page 6-87 was determined? For normal operating conditions, how is the T_T assumed to vary as function of elevation?

DNC Response
The WCNOC Question 6 response in Westinghouse LTR-SGMP-09-100-P (Enclosure 6) provides the response to this question.

3.0 DNC CHANGES RESULTING FROM RAI RESPONSES

As described in the response to RAI Question 24, a change was made to increase the leak rate factor for MPS3 from 2.03 to 2.49. The leak rate factor is applied to the operational leak rate to determine the accident leakage due to tube flaws contained within the tubesheet. The basis for the leak rate factor change was to ensure the accident leakage from a feedwater line break (FLB) accident, when it is assumed to be a heat-up event, remains bounded by the site accident induced leakage limit of 1.0 gpm at room temperature (gpmRT). The increased leak rate factor resulted in changes to the proposed reporting requirements in TS 6.9.1.7. This change is reflected in the marked-up TS pages for TS 6.9.1.7 (Enclosure 2) and the MPS3 regulatory commitments (Enclosure 3).

The accident induced leak rate limit is 1.0 gpm. The TS operational leak rate is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB leak rate ratio is only 2.49 resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

The revised leak rate factor did not affect the structural H* analysis because the H* structural analysis is bounded by normal operating conditions and not by accident conditions. The leak rate factor was not used in the structural H* analysis and there was no change to the normal operating conditions as previously evaluated; therefore, the H* length remains unchanged.

Based on the above information and a review of the enclosures, the RAI responses clarified the contents of Part A of this enclosure, but did not expand the scope of the LAR as originally evaluated and did not impact the conclusions of the SHC included in Part A. While the conclusion was unchanged, the SHC was impacted by the consideration of the FLB as a heat-up event. The revised SHC basis incorporates the change in leak rate factor from 2.03 to 2.49 that was driven by the additional design basis accident consideration. (Enclosure 4)

4.0 REFERENCES:

1) WCAP-17071-P, Revision 0, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," Revision 0, dated April 2009.

2) Letter dated July 10, 2009, from D. Wright, USNRC, to M. J. Ajluni, Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2, Request for Additional Information Regarding Steam Generator Program (TAC NOS. ME1339 and ME1340)."

3) Letter dated August 5, 2009, D. Wright, USNRC, to M. J. Ajluni, Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2, Request for Additional Information Regarding Steam Generator Program (TAC NOS. ME1339 and ME1340)."

4) Letter dated August 11, 2009, B. K. Singal, USNRC, to R. A. Muench, Wolf Creek Nuclear Operating Corporation; "Wolf Creek Generating Station – Request for Additional Information Regarding the Permanent Alternate Repair Criteria License Amendment Request (TAC No. ME1393)."

5) Letter NL-09-1265 dated August 28, 2009, M. J. Ajluni, Southern Nuclear Operating Company, Inc., to USNRC, "Vogtle Electric Generating Plant, Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria.

6) Letter NL-09-1375 dated August 28, 2009, M. J. Ajluni, Southern Nuclear Operating Company, Inc., to USNRC, "Vogtle Electric Generating Plant, Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification (TS) Sections 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" for Permanent Alternate Repair Criteria.

PART C

DISCUSSION OF CHANGE FROM PARC TO ONE-TIME ARC

1.0 SUMMARY DESCRIPTION

On September 2, 2009, in a teleconference between the NRC staff and industry personnel, the NRC staff indicated their concerns with eccentricity of the tubesheet tube bore in normal and accident conditions (RAI Question 4 of the July 10, 2009 letter (Part C, Reference 1) and RAI Question 1 of the August 5, 2009 letter (Part C, Reference 2)) have not been completely resolved. The staff further indicated that there was insufficient time to resolve these issues to support approval of the PARC amendment request for the fall 2009 refueling outages. The staff concluded, however, that the information submitted to date was sufficient to support a one-time TS amendment to provide sufficient time to resolve the eccentricity concerns. As such, DNC is proposing changes to TS 6.8.4.g and TS 6.9.1.7 to be a one-time change for MPS3 Refueling Outage 13 (3R13) and the subsequent operating cycle.

At MPS3, tube flaw indications within the tubesheet have only been found at the tube ends. Approximately 31,295 tube ends were inspected at MPS3 during 3R12 (fall 2009). In total, indications have been found in 146 tubes within 1-inch of the tube end. Of these, only 23 indications met the tube repair criteria in the current TSs using the interim alternate repair criteria (IARC). While no flaws in bulges/overexpansions within the tubesheet have been found at MPS3, a separate inspection program for these flaws has been implemented. This inspection program is in accordance with MPS3 current TSs and industry guidance.

Based on these inspections, a limited number of flaws exist in the tubesheets of MPS3 steam generators. The flaws that have been found are associated with residual stress conditions at the tube ends (over 20 inches below the top of the tubesheet). No indications of a 360 degree severance have been detected in any steam generator at MPS3. Consequently, the level of degradation in the MPS3 steam generators is very limited compared to the assumption of "all tubes severed at the depth of 13.1 inches below the top of the tubesheet" that was utilized in the development of the permanent H*. Therefore, structural integrity will be assured for the operating period between inspections allowed by the proposed one-time change to TS 6.8.4.g, "Steam Generator (SG) Program."

From a leakage perspective, projections of accident-induced steam generator tube leakage are based on leakage rate factors applied to leakage detected during normal operation. The multiplication factor of 2.49 used for MPS3 bounds the expected increased leakage in the event of an accident. The projected accident-induced leakage remains the same for both the one-time ARC and the PARC. No primary-to-secondary steam generator tube leakage has been detected during the current operating cycle at MPS3.

For MPS3, the number of tubes identified with flaws within the tubesheet is small in comparison to the input assumptions used in the development of the permanent H*. Consequently, significant margin exists between the current state of the MPS3 steam generators and the conservative assumptions used as the basis for the permanent H*. Structural and leakage integrity will continue to be assured for the operating period between inspections allowed by TS 6.8.4.g, "Steam Generator (SG) Program," with implementation of the proposed one-time H*.

Based upon the summary above, the SHC evaluation provided in Enclosure 4 was determined to bound the application for a one-time TS amendment. No additional revision to the SHC was required to support the one-time change.

TS 6.8.4.g and TS 6.9.1.7 required additional revision to support the change from a PARC to a one-time ARC. The marked-up TS pages in Enclosure 2 reflect all of the changes driven by Parts A, B and C of this enclosure. Additionally, the change from a PARC to a one-time ARC did impact the regulatory commitments associated with this LAR. The regulatory commitments, in final form, are included as Enclosure 3.

2.0 REFERENCES

1) Letter dated July 10, 2009, from D. Wright, USNRC, to M. J. Ajluni, Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2, Request for Additional Information Regarding Steam Generator Program (TAC NOS. ME1339 and ME1340)."

2) Letter dated August 5, 2009, D. Wright, USNRC, to M. J. Ajluni, Southern Nuclear Operating Company, Inc., "Vogtle Electric Generating Plant, Units 1 and 2, Request for Additional Information Regarding Steam Generator Program (TAC NOS. ME1339 and ME1340)."

ENCLOSURE 2

MARKED-UP TECHNICAL SPECIFICATION CHANGES FOR ONE-TIME ALTERNATE REPAIR CRITERIA

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

6.8.A g. 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 a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

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

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

PROCEDURES AND PROGRAMS (Continued)

Leakage is not to exceed 500 gpd per SG.

3. The operational LEAKAGE performance criterion is specified in RCS LCO 3.4.6.2, "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:

INSERT A

1. For MPS3 Refueling Outage 12 and the subsequent operating cycle, tubes with flaws having a circumferential component less than or equal to 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet do not require plugging. Tubes with flaws having a circumferential component greater than 203 degrees found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet shall be removed from service.

Tubes with service-induced flaws located within the region from the top of the tubesheet to 17 inches below the top of the tubesheet shall be removed from service. Tubes with service-induced axial cracks found in the portion of the tube below 17 inches from the top of the tubesheet do not require plugging.

When more than one flaw with circumferential components is found in the portion of the tube below 17 inches from the top of the tubesheet and above 1 inch from the bottom of the tubesheet with the total of the circumferential components greater than 203 degrees and an axial separation distance of less than 1 inch, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.

When one or more flaws with circumferential components are found in the portion of the tube within 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When one or more flaws with circumferential components are found in the portion of the tube within 1 inch

INSERT A

For Refueling Outage 13 and the subsequent operating cycle, 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.

PROCEDURES AND PROGRAMS (Continued)

~~from the bottom of the tubesheet and within 1 inch axial separation distance of a flaw above 1 inch from the bottom of the tubesheet, and the total of these circumferential components exceeds 94 degrees, then the tube shall be removed from service. When the circumferential components of each of the flaws are added, it is acceptable to count the overlapped portions only once in the total of circumferential components.~~

INSERT B

d. Provisions for SG tube inspections: Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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.

portions of the SG tube not excluded above,

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

INSERT B

For Refueling Outage 13 and the subsequent operating cycle, portions of the tube below 13.1 inches below the top of the tubesheet are excluded from this requirement.

FOR INFORMATION ONLY

PROCEDURES AND PROGRAMS (Continued)

associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

h. 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.

c. Requirements for (i) determining the unfiltered air inleakage 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.

The following are exceptions to Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0:

1. Appropriate application of ASTM E741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report, and

2. Vulnerability assessments for radiological, hazardous chemical and smoke, and emergency ventilation system testing were completed as documented in the UFSAR and other licensing basis documents. The exceptions to the Regulatory Guides (RG) referenced in RG 1.196 (i.e., RG 1.52, RG 1.78, and

6.9.1.6.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.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

STEAM GENERATOR TUBE INSPECTION REPORT

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

INSERT C

i. ~~Following completion of an inspection performed in Refueling Outage 12 (and any inspections performed in the subsequent operating cycle), the number of indications and location, size, orientation, whether initiated on primary or secondary side for each service-induced flaw within the thickness of the tubesheet,~~

INSERT C

i. During Refueling Outage 13 and the subsequent operating cycle, 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. During Refueling Outage 13 and the subsequent operating cycle, 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 2.49 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined; and

k. During Refueling Outage 13 and the subsequent operating cycle, the results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

STEAM GENERATOR TUBE INSPECTION REPORT (Continued)

and the total of the circumferential components and any circumferential overlap below 17 inches from the top of the tubesheet as determined in accordance with TS 6.8.4.g.c,

j. Following completion of an inspection performed in Refueling Outage 12 (and any inspections performed in the subsequent operating cycle), the primary-to-secondary LEAKAGE rate observed in each steam generator (if it is not practical to assign leakage to an individual SG the entire primary-to-secondary LEAKAGE should be conservatively assumed to be from one steam generator) during the cycle preceding the inspection which is the subject of the report, and

k. Following completion of an inspection performed in Refueling Outage 12 (and any inspections performed in the subsequent operating cycle), the calculated accident leakage rate from the portion of the tube below 17 inches from the top of the tubesheet for the most limiting accident in the most limiting steam generator.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, one copy to the Regional Administrator Region I, and one copy to the NRC Resident Inspector, within the time period specified for each report.

6.10 Deleted.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR Part 20:

ENCLOSURE 3

LIST OF REGULATORY COMMITMENTS FOR ONE-TIME ALTERNATE REPAIR CRITERIA

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

List of Regulatory Commitments For
One-Time Alternate Repair Criteria

The following table identifies those actions committed by Dominion Nuclear Connecticut, Inc. (DNC) for Millstone Power Station Unit 3 in support of the one-time alternate repair criteria for Refueling Outage 13 (3R13) and the subsequent operating cycle. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Due Date/Event	Commitment Type	
		One-Time Action (Yes/No)	Programmatic Action (Yes/No)
DNC commits to monitor for tube slippage as part of the steam generator (SG) tube inspection program.	Starting with 3R13 and during subsequent SG inspections.	Yes	No
DNC commits to perform a one-time verification of the tube expansion to locate any significant deviations in the distance from the top of tubesheet to the bottom of the expansion transition (BET). If any significant deviations are found, the condition will be entered into the plant's corrective action program and dispositioned. Additionally, DNC commits to notify the NRC of significant deviations.	Prior to startup following 3R13.	Yes	No
DNC commits to the following: For the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.49 and added to the total accident leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.49 and compared to the observed operational leakage. An administrative limit will be established to not exceed the calculated value.	At each scheduled inspection required by TS 6.8.4.g, "Steam Generator (SG) Program," beginning with 3R13.	Yes	No

Serial No: 09-525
Docket No. 50-423

ENCLOSURE 4

SIGNIFICANT HAZARDS CONSIDERATION FOR ONE-TIME ALTERNATE REPAIR CRITERIA

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

Significant Hazards Consideration For One-Time Alternate Repair Criteria

Dominion Nuclear Connecticut, Inc. (DNC) 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 steam generator inspection criteria and the steam generator inspection 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 steam generator tube inspection and repair criteria are the steam generator tube rupture (SGTR) event and the feedline break (FLB) postulated accidents.

During the SGTR event, the required structural integrity margins of the steam generator tubes and the tube-to-tubesheet joint over the H* distance will be maintained. Tube rupture in tubes with cracks within the tubesheet is precluded by the constraint provided by the tube-to-tubesheet joint. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet, and from the differential pressure between the primary and secondary side. 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," 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 integrity of the steam generator tubes and does not affect other systems, structures, components, or operational features. Therefore, the proposed change results in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. However, primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed changes since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter. Therefore, the proposed changes do not result in a significant increase in the consequences of a SGTR.

The consequences of a steam line break (SLB) are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the steam generator creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., a SLB) is limited by flow restrictions. These restrictions result from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications.

The leakage factor of 2.49 for Millstone Power Station Unit 3 (MPS3), for a postulated SLB/FLB, has been calculated as shown in Table RAI24-2 of Enclosure 5. The leakage factor of 2.49 is a bounding value for all steam generators, both hot and cold legs, in Table RAI24-2. Specifically, for the condition monitoring (CM) assessment, the component of leakage from the prior cycle from below the H* distance will be multiplied by a factor of 2.49 and added to the total leakage from any other source and compared to the allowable accident induced leakage limit. For the operational assessment (OA), the difference in the leakage between the allowable accident induced leakage and the accident induced leakage from sources other than the tubesheet expansion region will be divided by 2.49 and compared to the observed operational leakage.

The probability of a SLB is unaffected by the potential failure of a steam generator tube as the failure of the tube is not an initiator for a SLB event. SLB leakage is limited by leakage flow restrictions resulting from the leakage path above potential cracks through the tube-to-tubesheet crevice. The leak rate during postulated accident conditions (including locked rotor) has been shown to remain within the accident analysis assumptions for all axial and or circumferentially orientated cracks occurring 13.1 inches below the top of the tubesheet. The accident induced leak rate limit is 1.0 gpm. The technical specification (TS) operational leak rate is 150 gpd (0.1 gpm) through any one steam generator. Consequently, there is significant margin between accident leakage and allowable operational leakage. The SLB/FLB leak rate ratio is only 2.49 resulting in significant margin between the conservatively estimated accident leakage and the allowable accident leakage (1.0 gpm).

Therefore, 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 steam generator inspection criteria and the steam generator inspection 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 accident previously evaluated.

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

Response: No

The proposed change that alters the steam generator inspection criteria and the steam generator inspection reporting criteria maintains the required structural margins of the steam generator tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97-06, Revision 2, "Steam Generator Program Guidelines" and RG 1.121, are used as the bases in the development of the limited tubesheet inspection depth methodology for determining that steam generator tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the Nuclear Regulatory Commission (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 Electric Company, LLC (Westinghouse) report WCAP-17071-P, "H*: Alternate Repair Criteria for the Tubesheet Expansion Region in Steam Generators with Hydraulically Expanded Tubes (Model F)," 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 provides for large margins between calculated and actual leakage values in the proposed limited tubesheet inspection depth criteria.

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

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. WCAP-17071-P 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 a 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 NRC'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.