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Dominion®

May 8, 2008

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

Serial No.: 08-0249
NSSL/MAE R0
Docket No.: 50-423
License No.: NPF-49

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3
PROPOSED LICENSE AMENDMENT REQUEST, INTERIM ALTERNATE REPAIR CRITERIA (IARC) FOR STEAM GENERATOR (SG) TUBE REPAIR

Pursuant to 10 CFR 50.90, Dominion Nuclear Connecticut, Inc. (DNC) hereby requests an amendment in the form of changes to the Technical Specifications (TS) to Facility Operating License NPF-49 for Millstone Power Station Unit 3 (MPS3). This amendment proposes a one cycle revision to the MPS3 TS. Specifically, TS 6.8.4.g, "Steam Generator (SG) Program," and TS 6.9.1.7, "Steam Generator Tube Inspection Report," will be revised to incorporate an interim alternate repair criterion into the provisions for SG tube repair for use during the MPS3 2008 fall refueling outage (3R12) and the subsequent operating cycle (Cycle 13).

This license amendment request is based upon similar requests submitted by Wolf Creek, dated February 8, 2008, Vogtle 1 and 2, dated February 13, 2008 and Braidwood 1 and 2, dated February 25, 2008. As part of their review of the three submittals, the NRC issued requests for additional information (RAIs) which included, in aggregate, 17 questions. The utilities drafted the responses to Questions 1-5 and Westinghouse developed responses to Questions 6-17. These RAI responses were submitted to the NRC by Wolf Creek on March 21, 2008, Vogtle 1 and 2 on March 21, 2008 and Braidwood 1 and 2 on March 27, 2008. These RAI responses have been incorporated in this DNC license amendment request.

Enclosure 1 provides the discussion of the proposed change including incorporation of RAI responses. Enclosure 2 provides the marked-up versions of the proposed TS pages. Enclosure 3 contains supporting technical information and responses to RAI questions 6-17 provided by Westinghouse Electric Company LLC (Westinghouse), information that is proprietary to Westinghouse. Therefore, this information is supported by affidavits, signed by Westinghouse, the owner of the information. The affidavits set forth the bases on which the information may be withheld from public disclosure by the Commission and addresses 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 2.390 of the Commission's regulations. The affidavits are included in Westinghouse authorization letters CAW-08-2419 and CAW-08-2420, "Application for Withholding Proprietary Information from Public Disclosure", which also includes Proprietary Information Notices and Copyright Notices. The Westinghouse authorization letters are provided in Enclosure 5. 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 Enclosure 4.

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DNC has evaluated the proposed amendment and determined that it does not involve a significant hazards consideration as defined in 10 CFR 50.92. The basis for the determination is included in Enclosure 1. 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 September 4, 2008 to support the fall MPS3 refueling outage, which is currently scheduled to start in October 2008. Once approved, the amendment will be implemented prior to Mode 5 startup of MPS3.

DNC continues to remain engaged in industry activities associated with SG tube inspection and alternate repair criteria, both interim and permanent, for plants with thermally treated Alloy 600 tubes.

If you have any questions or require additional information, please contact Mr. Gary D. Miller at (804) 273-2771.

Sincerely,

Gerald T. Bischof
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 Gerald T. Bischof, 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 8th day of May, 2008.

My Commission Expires: August 31, 2008

Margaret B. Bennett
Notary Public

MARGARET B. BENNETT
Notary Public 354302
Commonwealth of Virginia
My Commission Expires Aug 31, 2008

Commitments in this letter:

For integrity assessments, the ratio of 2.5 will be used in completion of both the condition monitoring (CM) and operational assessment (OA) upon implementation of the IARC. For example, for the CM assessment, the component of leakage from the lower 4 inches of the most limiting steam generator during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage from the allowable limit during the limiting design basis accident minus the leakage from the other sources will be divided by 2.5 and compared to the observed leakage. An administrative limit will be established to not exceed the calculated value.

Enclosures:

1. Discussion of Change

2. Proposed Technical Specifications Pages (Marked-up)

3. Westinghouse Electric Company LLC Letters (Proprietary):

   Westinghouse Electric Company LLC, LTR-CDME-08-11 P-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated January 31, 2008.

   Westinghouse Electric Company LLC, LTR-CDME-08-43 P-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11 P-Attachment," dated March 18, 2008.

4. Westinghouse Electric Company LLC Letters (Non-Proprietary):

   Westinghouse Electric Company LLC, LTR-CDME-08-11 NP-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated January 31, 2008.

   Westinghouse Electric Company LLC, LTR-CDME-08-43 NP-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11-NP- Attachment," dated March 18, 2008.

   Westinghouse Electric Company LLC, LTR-CDME-08-25, Errata for LTR-CDME-08-11; "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated February 12, 2008.

   Westinghouse Electric Company LLC, LTR-CDME-08-97, "Applicability of LTR-CDME-08-11 and LTR-CDME-08-43 to Millstone Unit 3," dated April 25, 2008.

   Westinghouse LTR-CDME-08-107,– Applicability of LTR-CDME-08-11 and LTR-CDME-08-43 to Millstone 3 Uprate Conditions, May 5, 2008.

5. Westinghouse Electric Company LLC Authorization Letters:

   Westinghouse Electric Company LLC, CAW-08-2419, "Application for Withholding Proprietary Information from Public Disclosure," dated April 25, 2008.

   Westinghouse Electric Company LLC, CAW-08-2420, "Application for Withholding Proprietary Information from Public Disclosure," dated April 25, 2008.

cc: U.S. Nuclear Regulatory Commission
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ENCLOSURE 1

DISCUSSION OF CHANGE

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

DISCUSSION OF CHANGE

1.0 SUMMARY DESCRIPTION

This amendment proposes a one cycle revision to the Millstone Power Station Unit 3 (MPS3) Technical Specifications (TS) 6.8.4.g, "Steam Generator (SG) Program," and TS 6.9.1.7, "Steam Generator Tube Inspection Report," to incorporate an interim alternate repair criterion (IARC) into the provisions for SG tube repair criteria for use during the MPS3 2008 fall refueling outage and the subsequent operating cycle. This amendment application requests approval of an 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 (TTS) is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads. This amendment application is required to preclude unnecessary SG tube plugging while still maintaining tube structural and leakage integrity.

2.0 DETAILED DESCRIPTION

2.1 Proposed Change

The following specific changes to the MPS3 TS are proposed:

- TS 6.8.4.g – Steam Generator (SG) Program

  TS 6.8.4.g currently states:

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

  The criterion would be revised as follows, as noted in italic type:

  c. Provision 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.

Note: No need to change TS 6.8.4.g.d since MPS3 did not previously have a temporary ARC.

- TS 6.9.1.7 – Steam Generator Tube Inspection Report

  TS 6.9.1.7 currently states:

  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, and

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

TS 6.9.1.7 would be revised to add the following three additional reporting criteria:

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 SG) 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.

2.2 Background

TS 6.8.4.g requires a SG tube program be established and implemented to ensure SG tube integrity is maintained. SG tube integrity is maintained in meeting specified performance criteria (in TS 6.8.4.g) for structural and leakage integrity, consistent with the plant design and licensing bases. TS 6.8.4.g requires a condition monitoring assessment be performed during each outage during which the SG tubes are inspected to confirm the performance criteria are being met. TS 6.8.4.g also includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require that the inspection sample and methods of inspection be selected with the objective of detecting flaws of any type that may be present along the length of a 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. Tubes found by an inservice inspection to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged as specified in TS 6.8.4.g.

Reference 2, Westinghouse Letter LTR-CDME-08-11, "Interim Alternate Repair Criteria (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated January 31, 2008, provides the technical justification for an IARC requiring full-length inspection of the tubes within the tubesheet, but does not require plugging tubes if the extent of circumferential cracking observed in the region greater than 17 inches from the TTS is less than a value sufficient to permit the remaining circumferential ligament to transmit the limiting axial loads [the greater of 3 times the normal operating (NOP) loads or 1.4 times the steam line break (SLB) end cap loads]. Axial cracks below 17 inches from the TTS are not relevant to the tube pullout arguments because axial cracks do not degrade the axial load carrying capability of the tube. Axial cracks do not require plugging if they are below 17 inches from the TTS.

The limiting circumferential ligament has been defined by calculation. The calculation assumes that friction loads between the tube and tubesheet from any source are zero. This assumption avoids potential effects of uncertainties in tube and tubesheet material properties.

Also, based on the same assumption that the contact pressure between the tube and the tubesheet from any source is zero, the Reference 2 evaluation provides a basis for demonstrating the accident-induced leakage will always meet the value assumed in the plant's safety analysis if the observed leakage during normal operating conditions is within its allowable limits. The need to calculate leakage from individual cracks is avoided by the calculation of the ratio of accident-induced leakage to normal operating leakage.

3.0 TECHNICAL EVALUATION

An evaluation has been performed in Reference 2 to assess the need for removing tubes from service due to the occurrence of circumferentially or axially oriented cracks within the tubesheet. The primary conclusions of the evaluation are:

1. Axial cracks in tubes below a distance of 17 inches below the TTS are allowed to remain in service in the MPS3 SGs as they are not a concern relative to tube pullout and leakage capability.

2. Circumferentially oriented cracks in tubes below a distance of 17 inches below the TTS with an azimuthal extent of less than or equal to 203 degrees are allowed to remain in service for one cycle of operation (18-month SG tubing eddy current inspection interval).

3. Circumferentially oriented cracks in the bottom 1-inch of the tube or in the tube-to-tubesheet welds with an azimuthal extent of less than or equal to 94 degrees are allowed to remain in service for one cycle of operation (18-month SG tubing eddy current inspection interval).

A bounding analysis approach is utilized in the Reference 2 evaluation for both the minimum ligament calculation and the leakage ratio calculation. "Bounding" means that the most challenging conditions from the plants with hydraulically expanded Alloy 600TT tubing are used. For MPS3, the analysis bounds the current licensed power level as well as the stretch power uprate level (reference 21) proposed by DNC in the license amendment request dated July 13, 2007 (DNC Letter 07-0450). Three different tube diameters are represented by the affected plants (11/16" dia., Model F; 3/4" dia., Model D5; 7/8" dia., Model 44F). MPS3 has Model F SGs. The most limiting conditions for structural evaluation depend on tube geometry and applied normal operating loads; thus the conditions from the plant that result in the highest stress in the tube are used to define the minimum required circumferential ligament. The limiting leak rate ratio depends on the leak rate values assumed in the safety analysis and allowable normal operating leakage that results in the longest length of undegraded tube.

Questions Relating to Interim Alternate Repair Criteria for Steam Generator Tubes

This license amendment request is based upon similar requests submitted by Wolf Creek, dated February 8, 2008 (Reference 1), Vogtle 1 and 2, dated February 13, 2008 (Reference 15), and Braidwood 1 and 2, dated February 25, 2008 (Reference 19). As part of their review of the three submittals, the NRC issued requests for additional information (RAIs) which included, in

aggregate, 17 questions. The utilities drafted the responses to Questions 1-5 and Westinghouse developed responses to Questions 6-17. These RAI responses were submitted to the NRC by Wolf Creek on March 21, 2008 (Reference 3), Vogtle 1 and 2 on March 21, 2008 (Reference 14), and Braidwood 1 and 2 on March 27, 2008 (Reference 18). These RAI responses have been incorporated in this Dominion Nuclear Connecticut, Inc (DNC) license amendment request.

Discussion of Performance Criteria

The following NEI 97-06, Rev. 2 (Reference 6) performance criteria, which are included in MPS3's TSs, are the basis for the analyses documented in Reference 2:

The structural integrity performance criterion is:

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

The structural performance criterion is based on ensuring there is reasonable assurance a steam generator tube will not burst during normal operation or postulated accident conditions.

The accident-induced leakage performance criterion is:

> *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 SGs and leakage rate for an individual steam generator. Leakage is not to exceed 1 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.*

Primary-to-secondary leakage is a factor in the calculated dose due to releases outside containment resulting from a limiting design basis accident. The potential primary-to-secondary leak rate during postulated design basis accidents shall not exceed the offsite radiological dose consequences required by 10 CFR Part 100 guidelines or the radiological consequences to control room personnel required by GDC-19, or other NRC-approved licensing basis (e.g., 10 CFR 50.67).

The IARC for the tubesheet region have been developed to meet the above criteria. The structural criterion regarding tube burst is inherently satisfied because the constraint provided by the tubesheet to the tube prohibits burst.

Limiting Structural Ligament Discussion

As defined in Reference 2, the bounding structural ligament remaining which meets the NEI 97-06, Rev. 2, performance criterion described above and required for the tube to transmit the operational loads, is 126 degrees arc. This assumes the residual ligament is 100% of the tube wall in depth. A small circumferential initiating crack is predicted to grow to a through-wall condition before it is predicted to reach a limiting residual ligament. A residual ligament in a part-through-wall condition is not a significant concern, because the assumption that all circumferential cracks detected are 100% through-wall.

Consideration of Non Destructive Examination (NDE) Uncertainty

The NDE uncertainty must be addressed to assure the as-indicated circumferential arc of the reported crack is a reliable estimate of the actual crack. ETSS 20510.1 (Reference 7) describes the qualified technique used to detect circumferential primary water stress corrosion cracking (PWSCC) in the expansion transitions and in the tubesheet expansion zone (TEZ). The qualification data is provided in the ETSS.

The fundamental assumption for the IARC is that all circumferential cracks detected are 100% through-wall. Thus, even a shallow crack of small length will be considered to be through-wall. Further, tube burst is not an issue for the IARC because of the constraint provided by the tubesheet; rather, it is axial separation of the tube that is the principal concern. Assuming all circumferential cracks are through-wall reduces the inspection uncertainty to only the length of the cracks. Further, the accuracy of the length determination is an issue only when the indicated crack approaches the allowable crack length (the complement of the required residual ligament) and if the indicated crack length is a reasonable estimate of the structural condition of the tube.

Prior investigations have correlated the axial strength of the tube to the Percent Degraded Area (PDA) of the flaw (Reference 8). PDA takes into account the profile of the existing crack, including non-through-wall portions and shallow tails of the crack. Using the data from ETSS 20510.1 for cracks with a 90%, or greater, through-wall condition from both NDE and destructive examination, a comparison of the actual crack lengths and corresponding PDA for the cracks to a theoretical PDA which assumes that cracks are 100% through-wall has been made. All points with a PDA of 60% or greater fall below the theoretical PDA line. As the crack lengths increase, the separation of the actual PDA from the theoretical PDA tends to increase.

The conclusion that the as-indicated crack angle is conservative is further supported by considering the characteristics of the eddy current probes. Each probe has a "field of view," that is, a window of finite dimension in which it detects flaws. The field of view for the +Point probe typically varies between 0.1 inch to 0.2 inch depending on the specific characteristics of the probe. Therefore, as the probe traverses its path, a flaw will be detected as the leading edge of the field of view first crosses the location of the flaw, continuing until the trailing edge of the field of view passes the opposite end of the flaw. This is known as "lead-in" and "lead-out" of the probe and the effect of these are to render the indicated flaw length greater than the actual flaw length. Therefore, it is concluded that the indicated flaw length will be conservative relative to the actual flaw length, especially when assumed the entire length of the indicated flaw is 100% through-wall.

Based on the above, it is concluded that if the detected circumferential cracks are assumed to be 100% through-wall, the as-indicated crack lengths will be inherently conservative with respect to the structural adequacy of the remaining ligament. Therefore, no additional uncertainty factor is necessary to be applied to the as-measured circumferential extent of the cracks.

Consideration of Crack Growth

The growth of cracks due to PWSCC in this submittal request is dictated by four default growth rates from Reference 2. The distribution of growth rates is assumed to be lognormal. Typical values and conservative values are given, although it is recommended in Reference 9 to use the default values only when the historical information is not available and not to use the typical values unless the degradation is mild. (No significant crack growth data exists for the circumferential cracking in the tubesheet expansion region). Both sets provided in Reference 2 have mean values and 95% upper bound values. For this analysis, the typical 95% upper bound growth rate is used. The circumferential growth rates are expressed as inches per effective full power year (EFPY).

Table 1.0 Calculation of Required Minimum Ligament for 18 Months Operating Period

	Bounding Structural Ligament	EFPY (1)	Growth (In./EFPY) (2)	Growth (Deg./EFPY) (3)	Growth for Operating Period (degrees)	Minimum Structural Ligament (degrees)	Critical Ligament (degrees)
Tube	18 Calendar Month (CM) Operation	1.5	0.12	20.65	31	126	157
1) It is conservatively assumed that 1 EFPY= 1 Calendar Year 2) 95% upper value of typical growth rates from Reference 2 3) Based on smallest (Model F) mean tubesheet bore dimension							

The residual structural ligament must be adjusted for growth during the anticipated operating period between the current and the next planned inspection. For the MPS3 SGs, referring to Table 1.0 above, the maximum allowable through-wall circumferential crack size in a SG tube is 203° (= 360° – 157°) for one cycle of operation (18-month SG tubing eddy current inspection interval).

Note that the maximum allowable through-wall circumferential crack size in a SG tube was reduced to 203 degrees in the response to RAI Question 17 in Reference 4.

Primary-to-Secondary Leakage Discussion

A basis using the D'Arcy formula for flow through a porous medium is provided to assure the accident-induced leakage for the limiting accident will not exceed the value assumed in the safety analysis for the plant if the observed leakage during normal operation is within its limits for the bounding plant as discussed in Reference 2. The bounding plant envelopes all plants with recirculating SGs with Inconel 600 thermally treated tubes. The D'Arcy formulation was previously compared to other potential models such as the Bernoulli equation or orifice flow formulation and was found to provide the most conservative results. Assuming zero contact pressure in the tube joint, the length of undegraded crevice required to limit the accident-induced leakage to less than the value assumed in the safety analysis for the limiting plant is calculated to

be 3.78 inches. By definition of the IARC, a tube that can remain in service has an undegraded crevice of 17 inches. Therefore, a safety factor of 4.5 is available (17 inches /3.78 inches). Expressed in terms of length, the margin in the crevice is 13.22 inches. Significant margin on crevice length is available even if only the distance below the neutral axis of the tubesheet is considered. This distance is approximately 6.5 inches. A factor of safety of 1.72 is available. Expressed in terms of length, the margin in the crevice is 2.72 inches below the neutral axis of the tubesheet. During normal operating conditions, the tubesheet flexes due to differential pressure loads, causing the tubesheet holes above the neutral axis to dilate, and below the neutral axis, to constrict. No mechanical benefit is assumed in the analysis due to tubesheet bore constriction below the neutral axis of the tubesheet; however, first principles dictate the tubesheet bore and crevice must decrease. Therefore, the leakage analysis provided is conservative.

Based on the above, with a length of undegraded crevice of 17 inches, it is concluded that if the normal operating leakage is within its allowable value, the accident-induced leakage will also be within the value assumed in the MPS3 safety analysis. The total increase in leakage during a postulated accident condition would be less than a factor of 3.5 (0.35 gpm allowable leakage during a SLB event / 0.1 gpm allowable leakage during normal operating conditions).

For integrity assessments, the ratio of 2.5 will be used in completion of both the condition monitoring (CM) and operational assessment (OA) upon implementation of the IARC. For example, for the CM assessment, the component of leakage from the lower 4 inches of the most limiting steam generator during the prior cycle of operation will be multiplied by a factor of 2.5 and added to the total leakage from any other source and compared to the allowable accident analysis leakage assumption. For the OA, the difference in leakage from the allowable limit during the limiting design basis accident minus the leakage from the other sources will be divided by 2.5 and compared to the observed leakage. An administrative limit will be established to not exceed the calculated value.

Reporting Requirements

DNC proposes to report the following additional information associated with the IARC following the Fall 2008 inspections and any additional inspections during 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 TTS.

- The primary-to-secondary leakage rate observed in each SG (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 SG) during this cycle preceding the inspection which is the subject of the report.

- The calculated accident leakage rate from the portion of tube below 17 inches from TTS for the most limiting accident in the most limiting SG. A factor of 2.5 shall be used to relate this accident leakage to the related operational leakage.

The proposed reporting requirements are only required for the applicable period of the IARC.

Inspection and Repair of Tube

The tube below the IARC depth will be examined with a qualified technique, e.g., +Point probe. Axial flaws have no impact on the structural integrity of the tube in this region and may be left in service. Circumferential indications that exceed the maximum acceptable tube flaw size of 203 degrees will be plugged. The detection of flaws will result in sample expansion per EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines" (Reference 20). Stress concentration areas may be used to define the extent of the expansion, e.g., if a repairable indication is located in a bulge/overexpansion (BLG/OXP), the expansion may be limited to the non-inspected BLG/OXPs. The circumferential components of multiple flaws within 1 inch of each other axially will be combined in accordance with TSs 6.8.4.g.c.1. Furthermore, the circumferential component of flaws within the bottom 1 inch of the SG tubes is limited to 94 degrees.

Note: References and Tables in Section 5 of Reference 2 refer to the wrong section (e.g., Reference 6-1 should be 5-1). Westinghouse has issued an errata letter to correct the discrepancies, and a copy of the letter is provided in Enclosure 4.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Steam Generator (SG) tube inspection and repair limits are specified in Section 6.8.4.g, "Steam Generator (SG) Program" of the MPS3 Technical Specifications (TS). The current TS require that flawed tubes be repaired if the depths of the flaws are greater than or equal to 40 percent through-wall. The TS repair limits ensure that tubes accepted for continued service will retain adequate structural and leakage integrity during normal operating, transient, and postulated accident conditions, consistent with General Design Criteria (GDC) 14, 15, 30, 31, and 32 of 10 CFR 50, Appendix A. Specifically, the GDC state that the reactor coolant pressure boundary shall have "an extremely low probability of abnormal leakage ...and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDCs 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing ... to assess ... structural and leaktight integrity" (GDC 32). Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the steam generator tubing. Leakage integrity refers to limiting primary-to-secondary leakage during all plant conditions to within acceptable limits.

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

4.2 No Significant Hazards Consideration

DNC has evaluated whether 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

Of the various accidents previously evaluated, the proposed changes affect only the steam generator tube rupture (SGTR) event evaluation and the postulated steam line break (SLB), locked rotor, and control rod ejection accident evaluations. Loss-of-coolant accident (LOCA) conditions cause a compressive axial load to act on the tube. Therefore, since the LOCA tends to force the tube into the tubesheet rather than pull it out, it is not a factor in this amendment request. Another faulted load consideration is a safe shutdown earthquake (SSE); however, the seismic analysis of Model F steam generators has shown that axial loading of the tubes is negligible during an SSE.

At normal operating pressures, leakage from PWSCC below 17 inches from the TTS is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region.

For the SGTR event, the required structural margins of the steam generator tubes is maintained by limiting the allowable ligament size for a circumferential crack to remain in service to 203 degrees below 17 inches from the TTS for the subsequent operating cycle. Tube rupture is precluded for cracks in the hydraulic expansion region due to the constraint provided by the tubesheet. The potential for tube pullout is mitigated by limiting the allowable crack size to 203 degrees for the subsequent operating cycle. These allowable crack sizes take into account eddy current uncertainty and crack growth rate. It has been shown that a circumferential crack with an azimuthal extent of 203 degrees for the 18-month SG tubing eddy current inspection interval meets the performance criteria of NEI 97-06, Rev. 2, "Steam Generator Program Guidelines" and Draft Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." Therefore, the margin against tube burst/pullout is maintained during normal and postulated accident conditions and the proposed change does not result in a significant increase in the probability or consequence of a SGTR.

The probability of a SLB is unaffected by the potential failure of a SG tube as the failure of a tube is not an initiator for a SLB event. 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 and control rod ejection) has been shown to remain within the accident analysis assumptions for all axial or circumferentially oriented cracks occurring 17 inches below the top of the tubesheet. Since normal operating leakage is limited to 150 gpd (approximately 0.10 gpm), the attendant accident condition leak rate, assuming all leakage to be from indications below 17 inches from the top of the tubesheet, would be bounded by 0.35 gpm. This value is within the accident analysis assumptions for the limiting design basis accident for MPS3, which is the postulated SLB event.

Based on the above, the performance criteria of NEI-97-06, Rev. 2 and Draft Regulatory Guide (RG) 1.121 continue to be met and the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed change create the possibility of a new or different accident from any accident previously evaluated?

Response: No

The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. Tube bundle integrity is expected to be maintained for all plant conditions upon implementation of the interim alternate repair criteria. The proposed change does not introduce any new equipment or any change to existing equipment. No new effects on existing equipment are created nor are any new malfunctions introduced.

Therefore, based on the above evaluation, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No

The proposed change maintains the required structural margins of the steam generator tubes for both normal and accident conditions. NEI 97-06, Rev. 2 and RG 1.121 are used as the basis 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 staff for meeting GDC 14, 15, 31, and 32 by reducing the probability and consequences of an SGTR. RG 1.121 concludes that by determining the limiting safe conditions of tube wall degradation beyond which tubes with unacceptable cracking, as established by inservice inspection, should be removed from service or repaired, 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 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 in a tube or the tube-to-tubesheet weld, Reference 4 defines a length of remaining tube ligament that provides the necessary resistance to tube pullout due to the pressure induced forces (with applicable safety factors applied). Additionally, it is shown that application of the limited tubesheet inspection depth criteria will not result in unacceptable primary-to-secondary leakage during all plant conditions.

Based on the above, it is concluded that the proposed changes do not result in any reduction of margin with respect to plant safety as defined in the Updated Final Safety Analysis Report or bases of the plant Technical Specifications.

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

4.3 Precedents

Wolf Creek Nuclear Operating Corporation, Vogtle Electric Generating Plant Unit 1, and Braidwood Station Unit 2 were granted similar TS changes on April 4, April 9, and April 18, 2008, respectively. These changes modified the repair requirements for portions of the SG tubes greater than 17 inches below the top of the tubesheet.

4.4 Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance 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 with the implementation of the interim alternate repair criterion discussed above.

5.0 ENVIRONMENTAL CONSIDERATION

DNC has evaluated the proposed amendment for environmental considerations. The review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set 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. Letter from T. J. Garrett of Wolf Creek Nuclear Operating Corporation to USNRC dated February 8, 2008 (Serial No. ET 08-0009), "Docket No. 50-482: Revision to Technical Specification (TS) 5.5.9, 'Steam Generator (SG) Program' for Interim Alternate Repair Criteria."

2. Westinghouse Electric Company LLC letter, LTR-CDME-08-11, "Interim Alternate Repair Criteria (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated January 31, 2008.

3. Letter from T. J. Garrett of Wolf Creek Nuclear Operating Corporation to USNRC dated March 21, 2008 (Serial No. ET 08-0016), "Docket No. 50-482: Response to Request for Additional Information Related to License Amendment Request for an Interim Alternate Repair Criterion to Technical Specification 5.5.9, Steam Generator (SG) Program."

4. Westinghouse Electric Company LLC letter, LTR-CDME-08-43 P-Attachment "Response to NRC Request for Additional Information Relating to LTR-CDME-08-011 P-Attachment," dated March 18, 2008.

5. TSTF-449, Rev. 4, "Steam Generator Tube Integrity", Technical Specifications Task Force Standard Technical Specification Change Traveler, April 14, 2005.

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

7. ETSS #20510.1, Technique for Detection of Circumferential PWSCC at Expansion Transitions.

8. EPRI TR-107197, Depth Based Structural Analysis Methods for Steam Generator Circumferential Indications; November 1997.

9. EPRI 1012987, "Steam Generator Integrity Assessment Guidelines," July 2006.

10. NRC Letter, Wolf Creek Generating Station – Issuance of Amendment re: Revision to Technical Specification 5.5.9 on the Steam Generator Program (TAC No. MD8054), April 4, 2008.

11. Letter ET 08-0024, Docket No. 50-482: Supplemental Information Related to License Amendment Request for an Interim Alternate Repair Criterion to Technical Specification 5.5.9, "Steam Generator (SG) Program," dated March 30, 2008.

12. NRC Letter, Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Changes to Technical Specification (TS) Sections TS 5.5.9, "Steam Generator (SG) Program" and TS 5.6.10, "Steam Generator Tube Inspection Report" (TAC Nos. MD75450 and MD7451), April 9, 2008.

13. Letter NL-08-0522, Vogtle Electric Generating Plant Supplemental Information Related to License Amendment Request for an Interim Alternate Repair Criterion to Technical Specification 5.5.9, "Steam Generator (SG) Program," dated April 3, 2008.

14. Vogtle Electric Generating Plant Units 1 and 2, Response to Request for Additional Information Related to License Amendment Request to Revise Technical Specification (TS) "Steam Generator Tube Inspection Report" for Interim Alternate Repair Criterion, March 21, 2008.

15. Vogtle Electric Generating Station Plant Units 1 and 2, 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 Interim Alternate Repair Criterion, February 13, 2008.

16. NRC Letter, Braidwood Station, Units 1 and 2 – Issuance of Amendments re: Revision to Technical Specifications for the Steam Generator Program (TAC Nos. MD8158 and MD8159), April 18, 2008.

17. Exelon Letter RS-08-046, Supplemental Information Related to Steam Generator Tube Interim Alternate Repair Criteria Technical Specification, April 9, 2008.

18. Exelon Letter RS-08-031, Response to Request for Additional Information Regarding Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification, March 27, 2008.

19. Exelon Letter RS-08-016, Application for Steam Generator Tube Interim Alternate Repair Criteria Technical Specification Amendment, February 25, 2008.

20. EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6, October 2002.

21. Westinghouse LTR-CDME-08-107, Applicability of LTR-CDME-08-11 and LTR-CDME-08-43 to Millstone 3 Uprate Conditions, May 5, 2008.

Serial No. 08-0249
Docket No. 50-423

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATIONS PAGES

(MARKED-UP)

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

FOR INFO ONLY

May 31, 2007

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam Generator (SG) tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

NOTE

Separate ACTION entry is allowed for each SG tube.

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program:

1. Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days, and

2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. With required ACTION and associated completion time of ACTION a. not met or SG tube integrity not maintained:

1. Be in HOT STANDBY within 6 hours, and

2. Be in COLD SHUTDOWN within 36 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

FOR INFO ONLY

6.8.4

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."

INSERT A →

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.

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.

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

INSERT A - Insert as new item under 6.8.4.g.c on page 6-17b

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.

PROCEDURES AND PROGRAMS (Continued)

evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

6.8.5 Written procedures shall be established, implemented and maintained covering Section I.E, Radiological Environmental Monitoring, of the REMODCM.

6.8.6 All procedures and procedure changes required for the Radiological Environmental Monitoring Program (REMP) of Specification 6.8.5 above shall be reviewed by an individual (other than the author) from the organization responsible for the REMP and approved by appropriate supervision.

Temporary changes may be made provided the intent of the original procedure is not altered and the change is documented and reviewed by an individual (other than the author) from the organization responsible for the REMP, within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following 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, unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

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, ~~and~~

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

INSERT B

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.

INSERT B – Insert as new items in TS 6.9.1.7 on page 6-21

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.

ENCLOSURE 5

WESTINGHOUSE ELECTRIC COMPANY LLC AUTHORIZATION LETTERS:

- **Westinghouse Electric Company LLC, CAW-08-2419, "Application for Withholding Proprietary Information from Public Disclosure," dated April 25, 2008.**
- **Westinghouse Electric Company LLC, CAW-08-2420, "Application for Withholding Proprietary Information from Public Disclosure," dated April 25, 2008.**

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION UNIT 3

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Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: CAW-08-2419

April 25, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-CDME-08-43 P-Attachment, "Response to NRC Request for Additional Information Relating to LTR-CDME-08-11 P-Attachment," dated March 3, 2008 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2419 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Dominion Connecticut.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2419, 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.

Very truly yours,

J.A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: Jon Thompson (NRC O-7E1A)

bcc: J. A. Gresham (ECE 4-7A) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L (letter and affidavit only)
G. W. Whiteman, Waltz Mill
H. O. Lagally, Waltz Mill
C. D. Cassino, Waltz Mill
J. T. Kandra, Waltz Mill
R. C. Grendys, ECE 560H

CAW-07-2419

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 25th day of April 2008

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

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

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

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

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

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-CDME-08-43 P-Attachment, "Response to NRC Request for Additional Information (RAI) Relating to LTR-CDME-08-11 P-Attachment," dated March 3, 2008 (Proprietary), for submittal to the Commission, being transmitted by Dominion Connecticut Application for Withholding Proprietary Information from Public Disclosure to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Millstone Unit 3 is expected to be applicable to other licensee submittals in support of implementing an interim alternate repair criterion (IARC) that requires a full-length inspection of the tubes within the tubesheet but does not require plugging tubes with a certain arc length of circumferential cracking below 17 inches from the top of the tubesheet.

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

(a) Provide documentation of the analyses, methods, and testing for the implementation of an interim alternate repair criterion for the portion if the tubes within the tubesheet of the Millstone Unit 3 steam generators.

(b) Assist the customer in obtaining NRC approval of the Technical Specification changes associated with the interim alternate repair criterion.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for the purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

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Our ref: CAW-08-2420

April 25, 2008

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-CDME-08-11 P-Attachment, "Interim Alternate Repair Criterion (ARC) for Cracks in the Lower Region of the Tubesheet Expansion Zone," dated January 31, 2008 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-08-2420 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Dominion Connecticut.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-08-2420, 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.

Very truly yours,

J.A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: Jon Thompson (NRC O-7E1A)

bcc: J. A. Gresham (ECE 4-7A) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L (letter and affidavit only)
G. W. Whiteman, Waltz Mill
H. O. Lagally, Waltz Mill
C. D. Cassino, Waltz Mill
J. T. Kandra, Waltz Mill
R. C. Grendys, ECE 560H

CAW-08-2420

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed before me
this 25th day of April, 2008

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

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

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

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

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