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PNP 2016-001

March 3, 2016

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

SUBJECT: License Amendment Request – Revision to the Requirements for Steam Generator Tube Inspections and Repair Criteria in the Cold Leg Tube Sheet Region

Palisades Nuclear Plant
Docket 50-255
Renewed Facility Operating License No. DPR-20

REFERENCES:

1. NRC letter to Entergy Nuclear Operations, Inc., *Palisades Nuclear Plant - Issuance of Amendment Re: Tubesheet Inspection Depth for Steam Generator Tube Inspections (TAC No. MD2125)*, dated May 31, 2007 (ADAMS Accession Number ML071420216)

2. Entergy Nuclear Operations, Inc., letter to NRC, PNP 2011-11, *License Amendment Request for Steam Generator Cold-Leg Tubesheet Inspection*, dated March 3, 2011 (ADAMS Accession Number ML110680342)

3. NRC letter to Entergy Nuclear Operations, Inc., *Palisades Nuclear Plant - Supplemental Information Needed for Acceptance of License Amendment Request for Steam Generator Cold-Leg Tubesheet Inspection (TAC No. ME5780)*, dated April 5, 2011 (ADAMS Accession Number ML110910558)

4. Entergy Nuclear Operations, Inc., letter to NRC, PNP 2011-035, *Withdrawal of License Amendment Request for Steam Generator Cold-Leg Tubesheet Inspection*, dated April 19, 2011 (ADAMS Accession Number ML111090424)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Nuclear Operations, Inc. (ENO) requests Nuclear Regulatory Commission (NRC) review and approval of a proposed license amendment to revise Renewed Facility Operating License DPR-20 for the Palisades Nuclear Plant (PNP). ENO proposes to revise Appendix A, Technical Specifications (TS), as they apply to the Steam

PROPRIETARY
Attachments 4 and 7 contain confidential information submitted under 10 CFR 2.390. Withhold from public disclosure. When separated from Attachments 4 and 7, the remainder of the submittal may be decontrolled.

Generator (SG) Program requirements in TS Section 5.5.8 and TS Section 5.6.8. The purpose of the change is to implement an alternate repair criteria (ARC), that invokes a C - Star inspection length (C*), on a permanent basis for the cold leg side of the PNP SGs' tube sheet. Corresponding requirements to expand cold leg tube-in-tubesheet inspection scope, dependent on ARC implementation, and to report tube slippage are also added.

A similar change to the PNP's TS was approved in operating license amendment number 225 for the hot leg side of the SGs' tube sheet (Reference 1). In addition, ENO previously submitted a license amendment request (LAR) to adopt an ARC, that implemented C*, for the cold leg side of the SGs' tube sheet (Reference 2). The NRC LIC-109 acceptance review determined that supplemental information was needed to accept the LAR for review and requested that the supplemental information be submitted within 14 days of April 5, 2011 (Reference 3). ENO determined that providing the supplemental information within the requested time period was not possible and withdrew the LAR (Reference 4).

The proposed change would establish an ARC for the SG cold leg tubes invoking an inspection length, C*, below the bottom of the cold leg expansion transition or top of the tube sheet, whichever is lower. As an ARC, tube flaws detected below the C* region would not require plugging. Allowing flaws below the C* region to remain in service is inconsequential to SG tube structural and leak rate margins during normal operation and postulated accident conditions. The reason for the change is to minimize unnecessary tube plugging in order to maximize reactor coolant flow and primary to secondary heat transfer rate margins, which support the primary coolant system heat removal safety function.

ENO has evaluated the proposed changes in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that this change involves no significant hazards consideration. Attachment 1 provides a description and assessment of the proposed TS changes. Attachment 2 provides a mark-up of the existing TS pages showing the proposed changes. Attachment 3 provides the revised TS pages reflecting the proposed changes.

The technical bases addressing steam generator tube structural and leakage integrity were completed by Westinghouse Electric Company, LLC (Westinghouse) and are provided in Attachment 4 (proprietary version) and Attachment 5 (non-proprietary version). Additional documentation supporting the technical bases is referenced in Attachment 1, and has been previously submitted to the NRC, and therefore is not being resubmitted as part of this license amendment request.

Attachment 7 (proprietary version) and Attachment 8 (non-proprietary version) provide the supplemental information requested by the NRC in Reference 3.

Attachment 6 and Attachment 9 provides the Westinghouse proprietary authorization affidavits CAW-10-2752 and CAW-16-4380 which support the proprietary nature of Attachment 4 and Attachment 7, respectively. The affidavits set forth the basis for which the information may be withheld from public disclosure by the NRC and addresses the specific considerations listed in paragraph (b)(4) of 10 CFR 2.390.

ENO requests approval of the proposed license amendment request, by March 3, 2017. The amendment will be implemented within 60 days of approval.

In accordance with 10 CFR 50.91, ENO is notifying the State of Michigan of this proposed license amendment by transmitting a copy of this letter and non-proprietary attachments to the designated state official.

Summary of Commitments

This letter identifies no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 3, 2016.

Sincerely,

AJV/jpm

Attachments:
1. Description and Evaluation of Requested Change
2. Proposed Changes to Palisades Plant Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications Pages
3. Page Change Instructions and Revised Pages for Palisades Plant Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications
4. Westinghouse Proprietary Class 2, SG-SGMP-10-4-P, Revision 1, *Palisades Cold Leg Tubesheet Inspection Depth, C**, February 2010
5. Westinghouse Non-Proprietary Class 3, SG-SGMP-10-4-NP, Revision 1, *Palisades Cold Leg Tubesheet Inspection Depth, C**, February 2010
6. Westinghouse Electric Company, Affidavit CAW-10-2752, *Application for Withholding Proprietary Information from Public Disclosure,* November 17, 2010
7. Westinghouse Proprietary Class 2, LTR-SGMP-15-88, Rev. 1 P-Attachment, *Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis,* February 23, 2016
8. Westinghouse Non-Proprietary Class 3, LTR-SGMP-15-88, Rev. 1 NP-Attachment, *Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis,* February 23, 2016
9. Westinghouse Electric Company, Affidavit CAW-16-4380, *Application for Withholding Proprietary Information from Public Disclosure*, February 23, 2016

cc: Administrator, Region III, USNRC
Project Manager, Palisades, USNRC
Resident Inspector, Palisades, USNRC
State of Michigan

ATTACHMENT 1

Description and Evaluation of Requested Change

21 pages follow

1.0 SUMMARY DESCRIPTION

Entergy Nuclear Operations, Inc. (ENO) requests amending the Renewed Facility Operating License DPR-20, Docket No. 50-255 for Palisades Nuclear Plant (PNP). This amendment proposes to revise Appendix A, Technical Specification (TS) Section 5.5.8, *Steam Generator (SG) Program*, to add alternate repair criteria (ARC), invoking a C-star (C*) inspection length, on a permanent basis for the cold leg side of the PNP SGs tube sheet. Corresponding provisions to expand SG cold leg tube inspection scope, dependent on ARC implementation, and to report tube slippage, are also proposed.

The proposed change would establish an ARC for the SG cold leg tubes invoking an inspection length, C*, below the bottom of the cold leg expansion transition or top of the tube sheet, whichever is lower. As an ARC, tube flaws detected below the C* region would not require plugging. Allowing flaws below the C* region to remain in service is inconsequential to SG tube structural integrity and leak rate margins during normal operation and postulated accident conditions. The purpose of the change is to minimize unnecessary tube plugging in order to maximize reactor coolant flow through the SGs thereby maintaining primary to secondary heat transfer rate margins, which support the reactor coolant system heat removal safety function. A similar change to the PNP TS was approved in amendment number 225, for the SG hot leg side of the tubesheet (Reference 4).

ENO proposes the following changes to the PNP TS:

a. Modify PNP SG tube repair criteria requirements in TS 5.5.8 by adding ARC that would allow a SG tube to remain in service with a flaw that was found during inservice inspection with a depth equal to or exceeding 40% of the nominal tube wall thickness on the SG cold leg side of the SG tubesheet if the flaw is located below the C* inspection length. The proposed cold leg side C* inspection length is 12.5 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, and is equal to the previously approved hot leg C* inspection length (Reference 4). This 12.5 inch inspection length for PNP's SGs was developed using methods and test data from the C* generic report for Combustion Engineering (CE) designed steam generators, WCAP-16208-P, *NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions,* Revision 1, May 2005, (Reference 3), as supplemented for PNP in Attachment 4 and Attachment 7.

b. Add a SG tube inspection provision in TS 5.5.8 that is applicable when the alternate repair criteria, proposed above, is implemented. This complements the proposed cold leg ARC by expanding the inspection population to 100% of the cold leg inservice tubes while limiting the inspection depth to the C* length of 12.5 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower.

c. Add a SG tube inspection reporting requirement to TS 5.6.8 for tube slippage monitoring, discovery, and corrective actions. This will provide defense in depth by continued monitoring to ensure that SG tube slippage is not occurring, as predicted by the C* analyses. Then, if unexpected slippage is discovered, the requirement ensures that a proper evaluation of the occurrence and corrective actions are taken.

The supporting analyses in Attachments 4 and 7 are specific to the PNP SG application. As a result, the proposed changes to the TS do not conform to the verbiage in NUREG-1432, *Standard Technical Specifications – Combustion Engineering Plants*. The verbiage of the proposed TS adheres to that previously approved in PNP TS amendment number 225 for the SG hot leg.

2.0 DETAILED DESCRIPTION

ENO proposes a revision to PNP TS 5.5.8. Specifically, the current SG tube repair criteria and corresponding SG tube inspection provisions, in TS 5.5.8c. and 5.5.8d., respectively, would be revised to add a new requirement. Administrative formatting changes on the effected pages are also included. The proposed changes are as follows:

a. A new TS 5.5.8c.2 is added as worded below:

> "2. Tubes found by inservice inspection to contain flaws within 12.5 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service."

b. A new TS 5.5.8d.5 is added as worded below:

> "5. When the alternate repair criteria of TS 5.5.8.c.2 are implemented, inspect 100% of the inservice tubes to the cold-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8c.2 every 24 effective full-power months, or one refueling outage, whichever is less."

c. A new TS 5.6.8i. is added as worded below:

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

d. In TS 5.5.8c.1, a period, that was omitted mistakenly, is added at the end of the existing paragraph.

e. In TS 5.5.8d.4, a period in two references to TS 5.5.8c.1 is deleted to more accurately reflect the reference.

f. TS 5.5.8 is reformatted, by indenting the paragraphs, to be consistent with subsections in TS 5.5.7.

3.0 BACKGROUND

As described in Section 1.1, *Background*, of Attachment 4, the Pressurized Water Reactors Owners Group (PWROG) program, for plants with CE supplied SGs with explosive tube expansions, provided recommended tubesheet region inspection lengths based on the generic WCAP-16208-P, Revision 1, *NDE Inspection Length for CE Steam generator Tubesheet Region Explosive Expansions,* report. This inspection length is referred to as C*. Following the completion of the Westinghouse report, applications for license amendments were submitted for several plants, including PNP. Those applications included additional plant specific information to supplement WCAP-16208-P. Concurrently, plant specific W*, F*, and H* analyses were completed and TS amendments issued for differing SG designs. Analyses applicability is provided in the table below:

Inspection Length	Westinghouse Analysis	Applicable SG Design	Inspection Length Sensitivity
C*	WCAP-16208	CE Explansion – Explosively Expanded Tubes	Leakage integrity dominates structural integrity with additional margin to RG 1.121 requirements
W*	WCAP-14797	WEXTEX Expansions Explosively Expanded Tubes	Structural integrity dominates leakage integrity with additional margin to RG 1.121 requirements
H*	WCAP-17091	Hydraulically Expanded Tubes	Leakage integrity and structural integrity are equally dominate with adequate margin to RG 1.121 requirements
F*	WCAP-14225	Partial Depth Hard Roll Expansion	Leakage integrity and structural integrity are equally dominate with additional margin to RG 1.121 requirements

As can be inferred from the above table, SG design determines what parameters dictate the inspection length. In the case of PNP's design, the C* length is dictated by parameters that impact the leak rate analysis.

The industry need for steam generator (SG) alternate repair criteria (ARC) license amendments, for plants with Alloy 600TT tubing, started with the initial findings at Catawba Unit 2, in the fall of 2004. Other nuclear plants with Alloy 600TT tubing had found crack-like indications in tubes within the tubesheet as well. Most of the indications were found in the tack expansion region near the tube-end welds and were a mixture of axial and circumferential primary water stress corrosion cracking. Over time, these cracks can be expected to become more and more extensive, necessitating more extensive inspections of the lower tubesheet region and more extensive tube plugging or repairs. Increasing the number of SG tubes plugged reduces the heat removal capability of the SGs, resulting in the potential for shortening the useful lifetime of the SGs. To avoid these impacts, the affected licensees and their contractor, Westinghouse, developed proposed alternative inspection and repair criteria applicable to the tubes in the lowermost region of the tubesheets. These criteria, for CE SGs, are referred to as the C* criteria. The C* distance is the minimum engagement distance between the tube and tubesheet, measured downward from the top of the tubesheet (TTS), that is proposed as needed to ensure the structural and leakage integrity of the TTS joints. The proposed C* alternate repair criteria would exclude the portions of tubing below the C* distance from inspection and plugging requirements, on the basis that flaws below the C* distance are not detrimental to the structural and leakage integrity of the TTS joints.

Requests for permanent ARC amendments were proposed for a number of plants as early as 2005. PNP was part of this population, receiving an ARC amendment for the hot leg portion of the SG tubesheet on May 31, 2007 (Reference 4). Subsequently, the U.S. Nuclear Regulatory Commission (NRC, or the Commission) staff identified a number of issues, associated with H* based amendment proposals and other proposals made in 2009, and was unable to approve H* amendments on a permanent basis pending resolution of these issues. The staff found it did have a sufficient basis to approve H* amendments on an interim (temporary) basis, based on the relatively limited extent of cracking existing in the lower tubesheet region at the time the interim amendments were approved. Since these early proposals, the NRC has resolved the previously identified H* issues and has recently approved permanent H* ARC amendments for Turkey Point Nuclear Generating Station, Units 3 and 4 (Reference 20), and Indian Point Nuclear Generating Unit No. 2 (Reference 12).

4.0 SUMMARY OF LICENSING BASIS ANALYSIS

Since the generic WCAP for C* was completed, numerous TS amendments have been approved for differing SG designs, some being permanent and some being on an interim bases. During the approval process for these amendments, several issues arose and were resolved. The most significant issues were associated with the H* submittals. Listed below is a timeline of historical documents that pertain to PNP's ARC C* license amendment request (LAR). These documents in addition to WCAP-16208-P and Westinghouse letters of C* applicability to PNP SGs form a basis for this current submittal.

On May 31, 2007, the NRC issued amendment 225 to ENO for PNP SG tubesheet inspection depth on a permanent basis (Reference 4). Amendment 225 provided an alternate SG tube repair criteria for the hot leg tubesheet region using an inspection length (C*). The methodology to determine C* was developed for CE plants with SG tubes that were expanded into the tubesheet with an explosive process called "explansion." Specifically, amendment 225 revised the alternate repair criteria by limiting the inservice inspection length and adding a corresponding requirement to expand cold leg tubesheet inspection scope dependent on ARC implementation in TS 5.5.8c.1 and TS 5.5.8d.4, respectively. The TS sections state:

> TS 5.5.8c.1
>
> "Tubes found by inservice inspection to contain flaws within 12.5 inches below the bottom of the hot leg expansion transition or top of the hot leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service."
>
> TS 5.5.8d.4
>
> "When the alternate repair criteria of TS 5.5.8c.1 are implemented, inspect 100% of the inservice tubes to the hot leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8c.1 every 24 effective full-power months, or one refueling outage, whichever is less."

Referenced in the LAR associated with the hot leg C* amendment 225 was PNP operating license amendment number 223, issued July 6, 2006 (Reference 8), which incorporated Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, *Steam Generator Tube Integrity*, Revision 4. TSTF-449 incorporated the Nuclear Energy Institute, NEI 97-06, *Steam Generator Program Guidelines* (Reference 1). Amendment number 223 changes include, in part, a modification to TS 3.4.13, *PCS Operational LEAKAGE*, and the addition of TS 3.4.17, *SG Tube Integrity*. The new program requirements, implemented with amendment number 223, became effective after the 2006 refueling outage.

On October 19, 2009, the NRC issued an amendment to Sequoyah (W*) for the remaining life of the SGs (Reference 11). At that time the remaining life of the SGs was approximately 3 years, which was a basis for the LAR approval due to issues identified during the review of other stations' ARC LAR submittals that were based on the H* analysis. The H* issues were associated with tubesheet bore displacement eccentricities required to maintain the tube-to-tubesheet contact pressure, which were not fully resolved when the NRC issued the Sequoyah license amendment. The safety evaluation stated;

> "The H* review also raised an issue with respect to the conservatism of the relationship for determining a uniform diameter change that would produce the same change to the average tube-to-tubesheet contact pressure as would the actual non-uniform diameter changes from the 3D finite element analysis, and whether the tubesheet bore displacement eccentricities are sufficiently limited such as to ensure that the tube-to-tubesheet contact is maintained around the entire tube circumference. Although this latter issue was not fully resolved in the H* review, the NRC staff concludes that there is sufficient conservatism embodied in the proposed W* distance to ensure acceptable margins against tube pullout for the remaining life of the SGs, which is approximately 3 years based on the scheduled replacement of the SGs in 2012. This conclusion is based on a qualitative assessment of the conservatisms in the analyses as discussed in section 3.6 and the limited number of reported indications of cracking on the cold leg side to the SGs in all SGs."

Note that the open NRC issue discussed in the Sequoyah (W*) amendment has since been resolved as documented in the *Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment RE: H* Alternate Repair Criteria for Steam Generator Tube Inspection and Repair (TAC No. MF2269)*, dated September 5, 2014 (Reference 12).

On March 3, 2011, PNP submitted a LAR for a permanent SG cold leg alternate repair criteria based on the C* analysis (Reference 17). The submittal was similar to the previously approved PNP amendment for hot leg tube-in-tubesheet region inspection depth dated May 31, 2007 (Reference 4). The bases for the similar submittal was that due to lower PCS operational and accident temperatures in the SG cold leg tube-in-tubesheet region, as compared to the hot leg region, the tubes are less susceptible to pressurized water stress corrosion cracking (PWSCC) and therefore bounded by the previously approved hot leg tube sheet ARC. PNP also cited as precedent ARC C* amendments for the San Onofre Nuclear Generating Station (SONGS) Unit 2 and Unit 3 hot and cold leg tube-in-tubesheet regions and the St Lucie Plant, Unit 2, and Waterford Unit 3 hot leg tube-in-tubesheet regions.

In the March 3, 2011 PNP LAR, the cold leg C* inspection distance proposed was calculated to be 12.5 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower. The proposed cold leg C* ARC inspection length of 12.5 inches was equal to the hot leg C* ARC inspection length in TS 5.5.8c.1. This length applied to each tube in the cold leg tubesheet region of the PNP SGs.

On March 14, 2011, the NRC issued an amendment to Vogtle (H*) on only an interim bases (Reference 18). The interim status was mostly due to a significant shortcoming in the "thick-shell" 3-D finite element model displacements that were applied to the tube-to-tubesheet interaction model. This was resolved by development of the "Square Cell" model but not in sufficient time for Vogtle to resubmit for a permanent H* ARC. As a result, the NRC approved only an interim ARC. The safety evaluation stated:

> "Subsequent analysis by industry to address the staff's concerns revealed that tubesheet bore eccentricity per se did not have a significant bearing on the outcome of the H* analysis. However, these analyses also revealed a significant shortcoming in how displacements from the 3-dimensional (3D) finite element model of the lower SG assembly were being applied to the tube-to-tubesheet interaction model, which was based on thick shell equations. The industry developed a new tube-to-tubesheet interaction model to address this shortcoming and the H* analyses were updated accordingly. This more recent background is discussed in more detail as part of the staff's technical evaluation in Section 4.0 of this safety evaluation. Details of these more recent analyses became available for staff's review too late to support applications for a permanent H* amendment in the spring or fall of 2011. For this reason, the subject amendment requested by SNC is for an interim H* amendment, applicable to Refueling Outage 15 and the subsequent operating cycle for Unit 1 and to Refueling Outage 14 and the subsequent operating cycle for Unit 2."

On April 5, 2011, the NRC requested ENO to provide supplemental information to support the acceptance review of the PNP LAR for the SG cold leg tubesheet region inspection depth (Reference 13). The NRC staff requested that PNP supplement the application to address the staff's requested information by April 19, 2011. The letter requested the following supplemental information;

> "The reason for the non-acceptance is that the application is not complete. It does not address the lessons learned from the H* and their effect, if any, on the C* results. In other recent "star" proposals that extended the original hot leg application to the cold leg, the licensee either addressed this issue in its original submittal or the NRC staff submitted requests for additional information. Given the history of this issue (i.e., factoring in the lessons learned from the H* review into W*, F* cold leg amendment requests), the licensee should include this in its submittal. As a reference, the NRC staff has provided the following two documents which provide similar information requested by the staff:
>
> - Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Steam Generator Tube Inspection Alternate Inspection Criteria (TAC Nos. ME5067 and ME5068), dated March 14, 2011 (ADAMS Accession No. ML110660264).
>
> - Sequoyah Nuclear Plant (SQN) – Unit 2 – Technical Specifications (TS) Change 09-02 – W* Alternate Repair Criteria (ARC) for Steam Generator (SG) Tubes Cold Leg, dated May 21, 2009 (ADAMS Accession No. ML091530343).
>
> The following information is needed for the acceptance review:
>
> 1. Please justify the adequacy of the proposed C* distance in light of the lessons learned from the H* review."

On April 19, 2011, PNP withdrew the LAR for a permanent SG cold leg alternate repair criteria based on a C* length, dated March 3, 2011 (Reference 19). PNP stated in the letter the following reason for withdrawing the LAR:

> "Westinghouse has indicated that it considers it unlikely that a review of the H* lessons learned would negatively affect the C* inspection distance. However, to provide the requested information, ENO and Westinghouse have determined that considerable effort

> will be required to complete an evaluation to supplement the LAR. The information cannot be provided by the requested April 19, 2011, date."

Subsequent to the LAR withdrawal, Westinghouse completed a review of H* lessons learned that are applicable to C* and determined that there is no negative affect on the C* inspection distance (Attachment 7).

On September 5, 2014, the NRC issued an amendment to Indian Point Nuclear Generating Unit No. 2 (H*) on a permanent bases (Reference 12). This amendment is relevant because it documents resolution of several H* issues that were unresolved at the time PNP submitted its cold leg SG ARC LAR. The safety evaluation stated:

> "Requests for permanent H* amendments were proposed for a number of plants as early as 2005. The U.S. Nuclear Regulatory Commission (NRC, or the Commission) staff identified a number of issues with these early proposals and in subsequent proposals made in 2009, and was unable to approve H* on a permanent basis pending resolution of these issues. The staff found it did have a sufficient basis to approve H* amendments on an interim (temporary) basis, based on the relatively limited extent of cracking existing in the lower tubesheet region at the time the interim amendments were approved. The technical basis for approving the interim amendments is provided in detail in the staff's safety evaluations (SEs) accompanying issuance of these amendments. The staff recently approved similar permanent H* amendments for other plants with Model 44F SGs, such as Turkey Point Nuclear Generating Station, Units 3 and 4 (Reference 4)."

> "On June 14 and 15, 2010, the NRC staff conducted an audit at the Westinghouse Waltz Mill Site (Reference 9). The purpose of the audit was to gain a better understanding of the H* analysis pertaining to eccentricity, to review draft responses to the NRC staff questions in Reference 8, and to determine which documents would need to be provided on the docket to support any future requests for a permanent H* amendment. Based on the audit, including review of pertinent draft responses to Reference 8, the staff concluded that eccentricity did not appear to be a significant variable affecting either average T/TS contact pressure at a given elevation or calculated values of H*. The staff found that the average contact pressure at a given elevation is primarily a function of average bore diameter change at that elevation associated with the pressure and temperature loading of the tubesheet. Accordingly, the staff concluded that no adjustment of computed average bore diameter change considered in the thick shell model is needed to account for eccentricities computed by the 3-D FEA. The material reviewed during the audit revealed that computed H* values from the reference analysis continued to be conservative when the eccentricity adjustment factor is not applied."

The above timeline documents resolution to NRC issues concerning the H* analysis and thereby reconfirms the use of the C* analysis as a licensing bases for the proposed ARC for PNP's SGs cold leg tube region. The C* analysis primary licensing basis documents applicable to PNP's C* amendment are:

Document Number	Document Title	Date	Notes
WCAP-16208-P Revision 1 (Reference 3)	*NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions*	May 2005	1
LTR-CDME-06-80-P Revision 1 (Reference 15)	*Palisades Tubesheet Inspection Depth*	May 2006	1
LTR-CDME-06-40-P Revision 1 (Reference 14)	*Comments on the Application of WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions" to the Palisades Nuclear Power Plant*	May 2006	2
LTR-CDME-07-22-P (Reference 16)	*Responses to NRC Requests for Additional Information Regarding the Application of WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions" to the Palisades Nuclear Power Plant*	February 2007	2
SG-SGMP-10-4-P Revision 1 (Attachment 4)	*Palisades Cold Leg Tubesheet Inspection Depth, C**	February 2010	1
LTR-SG-SGMP-15-88, Rev. 1 P-Attachment (Attachment 7)	*Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis*	December 2015	3

Notes:
1. Principle Licensing Basis document
2. NRC request for additional information (RAI) on C* Analysis
3. H* applicability to C* Analysis

In summary, the licensing basis for the PNP cold leg ARC C* length LAR is based on the same analysis documents that were used for the May 31, 2007 approval of the PNP permanent hot leg ARC amendment 225. This included the referenced PNP amendment 223 that adopted TSTF-449, *Steam Generator Tube Integrity*, and NEI 97-06, *Steam Generator Program Guidelines*. In 2011, ENO submitted an amendment for a PNP cold leg ARC C* which was later withdrawn due to NRC RAIs associated with ongoing issues with other plants' LARs using H* analyses. Due to these outstanding H* issues, the NRC was not approving permanent ARC LARs as noted by the Vogtle and Sequoyah ARC interim amendments. In 2014, the NRC was able to resolve the H* issues and approve permanent ARC amendments for Turkey Point and Indian Point. In 2015, ENO contracted Westinghouse to review the lessons learned from the H* issues, as requested by the NRC staff in 2011, and determined, based on PNP steam generator design and operating parameters, that the resolved H* analyses issues are judged not relevant to the C* analyses. Therefore, the PNP cold leg ARC C* length primary licensing basis documents consist of the hot leg ARC C* amendment basis documents as supplemented by the Westinghouse letters resolving NRC requests for additional information (RAIs), including H* analysis issue applicability, and derivation of the PNP cold leg C* length using the industry generic C* analysis.

5.0 TECHNICAL EVLAUATION

5.1 Design

PNP is a two-loop CE design plant operating with a hot leg temperature of 583^{0}F and a cold leg temperature of 537^{0}F. The two SGs currently installed are CE Model 2530 replacement generators that were placed into operation in the fall of 1990. The tube material is mill annealed Alloy (Inconel) 600 with a 0.75-inch outside diameter and a 0.042-inch tube wall thickness. Each SG has 8219 tubes. The tubes were expanded through the full depth of the tube sheet, 20.5 inches, using an explosive process (explansion). The resultant interference fit between each of the tubes and tubesheet provides structural integrity to resist tube pull-out, and a leak resistant boundary between the primary and secondary systems. A seal weld joins the tube end to the cladding on the primary face of the tubesheet. The tube bundle is held in place by a stainless steel structure that is comprised of horizontal lattice-type eggcrate supports, vertical straps and diagonal straps. PNP's tubesheet is constructed as follows: rows 1-18 are U-bends, rows 19-138 are square bends, and there are 195 columns.

Operating experience with previously installed similar model CE SGs identified the potential susceptibility to fretting wear at the bat wing locations, in the area around the center stay cylinder region. As a result, 308 tubes in SG 'A' and 309 tubes in SG 'B' were preventatively plugged prior to SG installation.

Since installation of the PNP replacement SGs, based on inservice inspection results, additional SG tubes have been plugged. The current total number of tubes plugged, following the October 2015 refueling outage, are 550 for SG 'A' and 451 for SG 'B,' and the current number of active tubes in SG 'A' and 'B' are 7669 and 7768, respectively. Note that Attachment 4 indicates the current limiting active tube count is 7846 tubes in the limiting SG, which was the active tube count following the 2009 refueling outage and was the active tube count at the time of the hot leg C* LAR submittal.

5.2 Inspection Practices/Results

SG tubes are an integral part of the primary coolant pressure boundary and serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. Because of the importance of SG tube integrity, periodic inservice inspections of the SG tubes are required and are completed as part of the TS 5.5.8 SG program. These inspections detect degradation in the tubes resulting from interaction with the SG operating environment. In addition, these inspections provide a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. Tubes with degradation that exceed the tube repair limits specified in the TS are removed from service by plugging. The TS provide the acceptance criteria related to the results of SG tube inspections.

The PNP SG program requires that a degradation assessment be performed prior to each refueling outage per NEI 97-06 (Reference 1), Electric Power Research Institute (EPRI) SG Integrity Assessment Guidelines (Reference 9), and the EPRI SG Examination Guidelines (Reference 10). The degradation assessment is done to ensure that the performance criteria of structural integrity, accident induced leakage, and operational leakage will continue to be met for the duration of the next operating cycle. This is accomplished by selecting steam generator tubes to be inspected based on past inspection results, existing and potential degradation mechanisms, relevant operating experience, qualified examination techniques, and SG program requirements. Data gathered during the inspection is used as input to the subsequent SG condition monitoring and operational assessments.

The PNP 2015 refueling outage SG tube inspections of the cold leg tubesheet included +Point^{TM} probe examinations of outer three tubes exposed to the annulus, three tubes around the stay cylinder in the cold leg, and all tubes in the four outer rows. The inspection length was from +3 inches to -2 inches referenced to the secondary faces of the tubesheets in both SGs. The percent of the tubes inspected out of the total SG cold leg tube population was 12%.

The 2012 and 2014 refueling outage SG tube inspections of the cold leg tubesheet included +Point^{TM} probe examinations of the four outer rows in the cold leg +3 inches to -2 inches referenced to the secondary faces of the tubesheets in both SGs. The percent of the tubes inspected out of the total SG cold leg tube population was 11.3% (2012) and 12.0% (2014), respectively.

There were no outside diameter stress corrosion cracking (ODSCC) or primary water stress corrosion cracking (PWSCC) flaws identified in the three outer rows during the 2006, 2007, 2009 and 2010 refueling outage inspections, and in the four outer rows during the 2012 and 2014 refueling outage inspections. These results indicate that there is no active damage mechanism in the areas examined in the cold leg tube sheets of the SGs.

5.3 Reporting Requirements

PNP is required under TS 5.6.8, *Steam Generator Tube Inspection Report*, to submit specific information to the NRC within 180 days after the reactor coolant system reenters Mode 4 following a SG tube inspection. These reporting requirements include the location, orientation (if linear), and measured size (if available) of service-induced indications. Including those found in the tubesheet region. The program inspection requirements are met through use of eddy current probes deployed in the SG tubesheet region that are fully capable of detecting axial and circumferential indications. Degraded tube indications that exceed the tube repair limits within the C* length of the tubesheet region are plugged upon detection. These inspection and report activities provide a verification that the PNP SG operating experience continues to be conservative relative to the assumptions made in this LAR.

As a supplement to the above current reporting requirements ENO is proposing to add TS section 5.5.8i. to report the results of monitoring for tube displacement (slippage). Then if slippage is discovered, the implications of the discovery and corrective actions shall be included in the report. This will provide defense in depth with regard to the current SG program reporting requirements and the WCAP 16208-P analyses that concluded, with safety factors applied, that axial displacement (slippage) will not occur under normal operating and accident conditions. This is achieved by periodic axial displacement monitoring to ensure that SG tube slippage has not occurred. Finally, if unexpected slippage is discovered this requirement ensures that a proper evaluation of the occurrence and corrective actions are taken to prevent recurrence.

5.4 Analysis

PNP's cold leg ARC and corresponding C* inspection length was determined based on the same methodologies used in PNP's hot leg ARC and corresponding C* inspection length with the following differences:

1. Effects of PNP's hot zero power cold leg temperature of 532°F were considered when defining the cold leg C* distance.

2. Industry operating experience, since PNP's hot leg ARC TS amendment, associated with other "star" (W*, H*) ARC submittals was reviewed for impact on PNP's cold leg C* length analysis (e.g. coefficient of thermal expansion, 3D finite element analysis (FEA) model refinements).

The hot leg C* analysis methodology, which determines a recommended NDE inspection length (C*) for the CE steam generator tubesheet region with explosive expansions, was created through a joint industry test program conducted by Westinghouse. The program results were documented in WCAP-16208-P (Reference 3). This report demonstrates, through a conservative test and analysis program, that the recommended C* length tube inspections would ensure that this portion of the primary coolant system maintains its integrity. Tube integrity means that the tubes are capable of performing their functions in accordance with the plant design and licensing basis. Tube integrity includes both structural integrity and leakage integrity for which NEI 97-06 provides acceptance criteria which are met by this analysis. Structural integrity refers to maintaining adequate margins against gross failure, rupture, and collapse of the SG tubes. Leakage integrity refers to limiting primary-to-secondary leakage during normal operation, plant transients, and postulated accidents to ensure that the radiological dose consequences are within acceptable limits. Specifically, NEI 97-06 acceptance criteria for the tube to tubesheet joints is that they must resist burst with an internal pressure of 3 x NODP (normal operating differential pressure) or 1.4 x MSLB (main steam line break) conditions, and they must maintain primary to secondary accident-induced leakage below one gpm/SG or less as defined by specific plant licensing bases.

The hot leg C* length was further refined by applying applicable NRC RAIs from previous C* amendment requests. This was documented in LTR-CDME-06-40-P, revision 1 (Reference 14). This resulted in an updated leakage based inspection length including tubesheet deflection and NDE corrections of 12.5 inches.

WCAP-16208-P generated empirical pullout load and leakage rate test data for a number of tube to tubesheet joint mock-up samples. The testing determined that the joint length needed to ensure that both the burst integrity and the leakage criteria are met was dominated in all cases by the threshold length required to meet the leakage criterion. Therefore, the leakage criterion defines the required cold leg tube-to-tubesheet joint C* inspection length and bounds the inspection length for the cold leg side pullout criterion.

Tube Structural Integrity (Pullout Load)

Tube burst is prevented for a tube with defects in the tubesheet region because of the constraint provided by the tubesheet. Therefore, tube pullout would be a prerequisite for tube burst under the limiting internal pressure conditions of NEI 97-06. WCAP-16208-P evaluated the minimum joint length required to preclude tube pull-out at a load of 3 x NODP, which bounds 1.4 x MSLB differential pressure. The structural integrity, first slip pullout based inspection distance of 5.25 inches is significantly less than the proposed inspection distance of 12.5 inches. Pull analytical methods were utilized to correct the empirical data for tubesheet deflection effects on both the joint strength and leakage resistance. A sensitivity study to variations in tube and tubesheet materials potential negative impact on structural integrity length was performed (Attachment 7) and it determined that the pullout length would not be expected to be affected by more than about 1 inch which, when added to 5.25 inches, is still significantly less than the proposed 12.5 inches.

Tube Leakage Integrity (Leak Rate)

The NEI 97-06 primary to secondary accident-induced leakage criteria of one gpm per SG exceeds the limiting condition of operation (LCO) and accident analysis leakage limits for PNP, which has a limit of 0.3 gpm per SG. To account for this disparity and to allow margin for other possible leak sources, WCAP-16208-P evaluated the minimum joint length required to maintain primary to secondary accident-induced leakage at 0.1 gpm per SG, assuming that 100% of the SG tubes were leaking below the C* depth.

The TSTF-449 submittal to the NRC, which was approved as PNP operating license amendment number 223, established the current PNP TS SG program that ensures tube integrity is maintained. In TS 3.4.13., *PCS Operational Leakage*, LCO 3.4.13, item d, states that operational leakage through any one SG shall be limited to 150 gallons per day (0.1 gpm). The UFSAR sections 14.14, *Steam Line Rupture Incident*, 14.15, *Steam Generator Tube Rupture with a Loss of Offsite Power*, and 14.16, *Control Rod Ejection*, leakage assumption is 0.3 gpm (432 gallons per day per steam generator). For the tube rupture accident, this 0.3 gpm leakage is in addition to the break flow rate associated with the rupture of a single SG tube. Therefore, the LCO leakage limit is conservatively less than the design basis accident-induced leakage limit.

An additional conservatism was introduced by assuming that 100% of the SG tubes are severed by a 360° circumferential crack immediately below the C* inspection length. Axial position uncertainties associated with eddy current examinations were also accounted for by adding a correction factor to the data. The final result of WCAP-16208-P for PNP was a C* value of 11.6 inches.

In 2006, the C* inspection distance was analyzed (Reference 14) for the SG hot leg to include additional allowances for T_{hot} temperature difference effects for PNP compared to the WCAP-16208 analysis, and for potential differences associated with use of first slip pullout loads compared to peak pullout loads of test specimens described in WCAP-16208-P. The re-calculated C* inspection distance including the above effects increased the inspection depth to 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower. Application of this analysis to the SG cold leg is included in Attachment 4.

The only difference in the calculation of the C* inspection depth for the cold leg of the SG tubesheet, as compared to the calculation of the C* inspection depth for the hot leg, is the lower temperature at the cold leg tube, tubesheet, and channelhead compared to the hot leg. To further verify application of the hot leg inspection depth to the cold leg tubesheet, a sensitivity study based on the lessons learned from previous H* submittals was conducted (Attachment 7) and it was concluded that potential lessons learned from the H* program are judged not relevant to the C* analysis. Therefore, the analyses (Attachment 4) that determined a 12.5 inch C* inspection length for the cold leg at PNP is conservative and technically supported.

5.5 Editorial Changes

The proposed changes of adding a period to TS 5.5.8c.1, deleting two periods from TS 5.5.8d.4 and indenting paragraphs in TS 5.5.8 are editorial and formatting corrections to PNP TS. They are included in this LAR to correct previous errors, since they occur in the same TS section affected by the proposed SG cold leg alternate repair criteria (C*) TS section changes.

6.0 REGULATORY ANALYSIS

6.1 Applicable Regulatory Requirements/Criteria

The proposed changes would continue to satisfy Regulatory Guide (RG) 1.121 (Reference 2), *Bases for Plugging Degraded PWR Steam Generator Tubes*, tube burst and collapse criteria. Operation of the SGs with potential tube degradation below the C* inspection length would continue to meet the historical safety margin guidance in RG 1.121.

Under the requirements of 10 CFR 50.65, *Requirements for monitoring the effectiveness of maintenance at nuclear power plants*, licensees classify SGs as risk significant components because they are relied on to remain functional during and after design basis events. The performance criteria in TS 5.5.8b. are used to demonstrate that the condition of the SG "is being effectively controlled through the performance of appropriate preventive maintenance" (10 CFR 50.65(a)(2)). Meeting the TS performance criteria that were incorporated from NEI 97-06, Rev. 3, *Steam Generator Program Guidelines*, provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining the primary coolant system pressure boundary.

NEI 97-06, and its referenced EPRI guidelines, define a SG program that provides the appropriate preventive maintenance that meets the intent of the 10 CFR 50.65. The SG performance criteria in NEI 97-06 are;

> *The structural integrity performance criterion is the following:*
>
> *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, 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 accident-induced leakage performance criterion is the following:*
>
> *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 gpm per steam generator, except for specific types of degradation at specific location when implementing alternate repair criteria as documented in the Steam Generator Program technical specifications.

The operational leakage performance criterion is the following:

The RCS [reactor coolant system] *operational primary-to-secondary, leakage through any one steam generator shall be limited to 150 gallons per day.*

The safety significant portion of a SG tube is the length of the tube that is required to maintain structural and leakage integrity over the full range of SG operating conditions and the most limiting accident conditions. Evaluations in the attached analyses have determined that, degradation in tubing below the safety significant portion (C* length) of the tube does not require plugging, and they serve as the bases for the tubesheet inspection program. As such the PNP SG inspection program provides a high level of confidence that the structural and leakage criteria are maintained during operating and accident conditions.

10 CFR 50, Appendix A, *General Design Criteria* [GDC] *for Nuclear Power Plants* – GDC 14, 30, and 32 define requirements for the reactor coolant pressure boundary with respect to structural and leakage integrity. SG tubing and tube repair constitutes a major fraction of the reactor coolant pressure boundary surface area. SG tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure. The SG program establishes performance criteria, repair criteria, repair methods, inspection intervals and the methods necessary to meet them. These requirements provide reasonable assurance that tube integrity would be maintained in the interval between SG inspections.

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

10 CFR 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,* establishes quality assurance criteria that contain pertinent requirements that apply to all activities affecting the safety related functions of these components. These requirements are described in criteria IX, *Control of Special Processes*, XI, *Test Control*, and XVI, *Corrective Action*.

10 CFR 100, *Reactor Site Criteria,* establishes 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 SG tubing may result in a challenge to containment and, therefore, involve an increased risk of radioactive release.

6.2 Precedent

On May 31, 2007, the NRC issued PNP permanent operating license amendment number 225 (Reference 4), which revised PNP TS to allow for a SG tube alternate repair criteria for the hot-leg tube-in-tubesheet region by applying the C* methodology. This license amendment was based on WCAP-16208-P and PNP plant specific analysis (References 3, 14, and 16). As described in this proposed LAR for the SG cold legs, the hot-leg C* analysis inspection distance of 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, is still conservative, given the H* lessons learned, and therefore this amendment is applicable to the PNP cold-leg submittal.

On October 19, 2009, the NRC issued Sequoyah Nuclear Plant (SQN) an interim operating amendment number 318, (Reference 11) which revised SQN technical specifications to allow for a SG tube alternate repair criteria for the cold legs tube-in-tubesheet region by applying the W* methodology. In the interim status of the amendment was partly due to issues raised associated with H* methodology and the fact the SQN planned to replace the SGs. The safety evaluation (SE) stated: *In summary, the NRC staff considered the issues identified during the H* review in the review of this proposal and concludes that there is sufficient conservatism embodied in the proposed W* distance to ensure acceptable margins against tube pullout for the remaining life of the SGs.* Since 2009 the H* review issues have been resolved as discussed in the below Indian Point Nuclear Generating Unit No. 2 license amendment dated September 5, 2014. The SQN precedent is applicable to PNP because the W* and C* analysis are both for SGs that were manufactured by explosive expansion of the SG tubes into the tubesheets and both are for a cold leg application.

On September 5, 2014, the NRC issued Indian Point Nuclear Generating Unit No. 2 a permanent operating amendment number 277 (Reference 12), which revised Indian Point Energy Center (IPEC) Technical Specifications to allow for a SG tube alternate repair criteria for the hot leg and cold legs tube-in-tubesheet region by applying the H* methodology. The IPEC precedent is applicable to PNP because like PNP it is for a cold leg application, was issued as a permanent amendment, and resolved the circa 2009 H* analysis issue. The SE states:

> *Requests for permanent H* amendments were proposed for a number of plants as early as 2005. The U.S. Nuclear Regulatory Commission (NRC, or the Commission) staff identified a number of issues with these early proposals and in subsequent proposals made in 2009, and was unable to approve H* amendments on a permanent basis pending resolution of these issues. The staff found it did have a sufficient basis to approve H* amendments on an interim (temporary) basis, based on the relatively limited extent of cracking existing in the lower tubesheet region at the time it interim amendments were approved. The technical basis for approving the interim amendments is provided in detail in the staff's safety evaluations (SEs) accompanying issuance of these amendments. The staff recently approved similar permanent H* amendments for other plants with Model 44F SGs, such as Turkey Point Nuclear Generating Station, Units 3 and 4 (Reference 4).*

As described in this proposed LAR, due to design differences associated with Combustion Engineering C* SGs versus Westinghouse H* SGs, the issues associated with previous H* submittals do not impact the C* analysis results.

6.3 <u>No Significant Hazards Consideration</u>

This amendment proposes to revise Appendix A, Technical Specification (TS) 5.5.8c, *Steam Generator (SG) Program, Provisions for SG tube repair criteria*, to add an alternate repair criteria (ARC), based on a C-star (C*) inspection length, on a permanent basis for the cold leg side of the PNP SGs' tube sheet. A corresponding requirement to expand cold leg tubesheet inspection scope is also added to TS 5.5.8d dependent on ARC implementation.

Entergy Nuclear Operations, Inc. (ENO) has evaluated the safety significance of the proposed amendment to the Palisades Nuclear Plant (PNP) Technical Specification (TS) 5.5.8, *Steam Generator (SG) Program* according to the criteria of 10 CFR 50.92, *Issuance of Amendment*. ENO has determined that the subject changes do not involve a significant hazards consideration, as discussed below:

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

Response: No.

Previously evaluated accidents are initiated by the failure of plant structures, systems, or components. The proposed change alters the SG cold leg repair criteria by limiting tube inspection length in the cold leg tubesheet, to the safety significant section, C* length, and, as such, does not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. Therefore the proposed change has no significant effect upon previously evaluated accident probabilities or consequences.

The proposed amendment to revise the PNP SG tube repair criteria in TS 5.5.8c, does not involve a significant increase in the probability of an accident previously evaluated. Alternate repair criteria are being proposed for the cold leg side of the SGs that duplicate the current alternate repair criteria for the hot leg side of the SGs, in TS Section 5.5.8c.1. The proposed SG tube inspection length maintains the existing design limits of the SGs and therefore does not increase the probability or consequences of an accident involving a tube rupture or primary to secondary accident-induced leakage, as previously evaluated in the PNP Updated Final Safety Analysis Report (UFSAR). Also, the Nuclear Energy Institute (NEI) *Steam Generator Program Guidelines* (NEI 97-06) performance criteria for structural integrity and accident-induced leakage, which are incorporated in PNP TS 5.5.8, would continue to be satisfied.

Implementing an alternate repair criteria would allow SG tubes with flaws below the C* length to remain in service. The potential consequences to leaving these flawed tubes inservice are tube burst, tube pullout, and accident induced tube leakage. Tube burst is prevented for a tube with defects within the tubesheet region because of the constraint provided by the tubesheet. Tube pullout could result from the axial forces induced by primary to secondary differential pressures that occur during the bounding event of the main steam line break. A joint industry test program report, WCAP-16208-P, *NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions*, Revision 1, May 2005, has defined the non-degraded tube to tubesheet joint length (C*) required to preclude tube pullout and maintain acceptable primary to secondary accident-induced leakage, conservatively assuming a 360 degree circumferential through wall crack exists immediately below this C* length.

The PNP UFSAR Sections 14.14, *Steam Line Rupture Incident*, 14.15, *Steam Generator Tube Rupture with a Loss of Offsite Power*, and 14.16, *Control Rod Ejection*, primary coolant system leakage limit is 0.3 gallon per minute (gpm) (432 gallons per day) in the unaffected SG. For the tube rupture accident, this 0.3 gpm leakage is in addition to the break flow rate associated with the rupture of a single SG tube. The WCAP-16208-P report used a primary to secondary accident-induced leakage criteria value of 0.1 gpm to derive the C* length. Use of 0.1 gpm ensures that the PNP TS limiting accident-induced leakage of 0.3 gpm is met.

For PNP, the derived C* length for the cold leg side of the SGs is 12.5 inches, which is the same C* length, as the current TS, for the hot leg side of the SGs. Any degradation below the C* length is shown by test results and analysis to meet the NEI 97-06 performance criteria, thereby precluding an increased probability of a tube rupture event or an increase in the consequences of a steam line rupture incident or control rod ejection accident.

Therefore, the C* lengths for the SG hot and cold legs provide assurance that the NEI 97-06 requirements for tube burst and leakage are met and that they conservatively derived maximum combined leakage from both tubesheet joints (hot and cold legs) is less than 0.2 gpm at accident conditions. This combined leakage criterion of 0.2 gpm in the faulted loop retains margin against the PNP TS allowable accident-induced leakage of 0.3 gpm per SG.

In summary, the proposed changes to the PNP TS maintain existing design limits, meet the performance criteria of NEI 97-06 and Regulatory Guide 1.121, and the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated in the UFSAR.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed amendment provides for an alternate repair criteria that excludes the lower portion of the steam generator cold leg tubes from inspection below a C* length by implementing an alternate repair criteria. It does not affect the design of the SGs or their method of operation. It does not impact any other plant system or component. Plant operation will not be altered, and all safety functions will continue to perform as previously assumed in the accident analysis.

The proposed amendment does not introduce any new equipment, change existing equipment, create any new failure modes for existing equipment, nor introduce any new malfunctions resulting from tube degradation. SG tube integrity is shown to be maintained for all plant conditions upon implementation of the proposed alternate repair criteria for the SG cold leg tubesheet region.

The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated because SG tube leakage limits and structural integrity would continue to be maintained during all plant conditions upon implementation of the proposed alternate repair criteria to the PNP TSs. The alternate repair criteria does not introduce any new mechanisms that might result in a different kind of accident from those previously evaluated. Even with the limiting circumstances of a complete circumferential separation (360 degree through wall crack) of a tube below the C* length, tube pullout is precluded and leakage is predicted to be maintained within the TS and accident analysis limits during all plant conditions.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change provides an alternate repair criteria for the SG cold leg that invokes a C* inspection length criteria. The proposed amendment does not involve a significant reduction in a margin of safety since design SG primary to secondary leakage limits have been analyzed to continue to be met. This will ensure that the SG cold legs tubes continue to function as a primary coolant system boundary by maintaining their integrity. Tube integrity includes both structural and leakage integrity. The proposed cold leg tubesheet inspection C* depth, of 12.5 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, would ensure tube integrity is maintained during normal and accident conditions because any degradation below C* is shown by test results and analyses to be acceptable.

Operation with potential tube degradation below the proposed C* cold leg inspection length within the tubesheet region of the SG tubing meets the recommendations of NEI 97-06 SG program guidelines. Additionally, the proposed changes also maintain the structural and accident-induced leakage integrity as required by NEI 97-06.

The total leakage from an undetected flaw population below the C* inspection length for the cold leg tubesheet under postulated accident conditions is accounted for, in order to assure it is within the bounds of the accident analysis.

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

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

6.4 Conclusion

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

7.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment would change a requirement with respect to installed facility components located within the restricted area of the plant as defined in 10 CFR 20. 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.

8.0 REFERENCES

1. Nuclear Energy Institute, NEI 97-06, Revision 3, *Steam Generator Program Guidelines*, dated January 2011 (ADAMS Accession Number ML111310708)

2. NRC, Regulatory Guide 1.121, *Basis for Plugging Degraded PWR Steam Generator Tubes*, dated August 1976 (ADAMS Accession Number ML003739366)

3. Westinghouse report, WCAP-16208-P, Revision 1, *NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions*, dated May 2005 (ADAMS Accession Number Non-Proprietary version ML051520417)

4. NRC letter to Entergy Nuclear Operations, Inc., *Palisades Nuclear Plant – Issuance of Amendment Re: Tubesheet Inspection Depth for Steam Generator Tube Inspections (TAC No. MD2125)*, dated May 31, 2007 (ADAMS Accession Number ML071420216)

5. Nuclear Management Company, LLC letter to NRC, *License Amendment Request Regarding Tubesheet Inspection Depth for Steam Generator Tube Inspections*, dated May 30, 2006 (ADAMS Accession Number ML061560407)

6. Nuclear Management Company, LLC letter to NRC, *Response to Request for Additional Information Regarding Proposed C* License Amendment Request for Steam Generator Tube Repair in the Tubesheet (TAC No. MD2125)*, dated February 27, 2007 (ADAMS Accession Number ML070640058)

7. Nuclear Management Company, LLC letter to NRC, *Response to Request for Additional Information Regarding Proposed C* License Amendment Request for Steam Generator Tube Repair in the Tubesheet (TAC No. MD2125)*, dated April 10, 2007 (ADAMS Accession Number ML071030330)

8. NRC letter to Nuclear Management Company, LLC, *Palisades Plant – Issuance of Amendment Re: Steam Generator Tube Integrity Technical Specifications (TAC No. MD0196)*, date July 6, 2006 (ADAMS Accession Numbers ML061880165 & ML061660197)

9. Electric Power Research Institute (EPRI) report 1019038, *Steam Generator Management Program: Steam Generator Integrity Assessment Guidelines*, Revision 3, dated October 2008 (ADAMS Accession Number ML100480243)

10. Electric Power Research Institute (EPRI) report 1013706, *Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines*, Revision 7, dated October 2007 (ADAMS Accession Number ML080450582)

11. NRC letter to Tennessee Valley Authority, *Sequoyah Nuclear Plant, Unit 2 – Issuance of Amendment to Allow Use of the W* Alternate Repair Criteria for Steam Generator Tubes (TS-09-02) (TAC NO. ME1343)*, dated October 19, 2009 (ADAMS Accession Number ML092810333)

12. NRC letter to Entergy Nuclear Operations, Inc., *Indian Point Nuclear Generating Unit No. 2 – Issuance of Amendment RE: H* Alternate Repair Criteria for Steam Generator Tube Inspection and Repair (TAC No. MF3369)*, dated September 5, 2014 (ADAMS Accession Number ML14198A161)

13. NRC letter to Entergy Nuclear Operations, Inc., *Palisades Nuclear Plant – Supplemental Information Needed for Acceptance of License Amendment Request for Steam Generator Cold-leg Tubesheet Inspection (TAC No. ME5780)*, dated April 5, 2011 (ADAMS Accession Number ML110910558)

14. Westinghouse letter to PNP, LTR-CDME-06-40-P, *Comments on the Application of WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions" to the Palisades Nuclear Power Plant*, Revision 1 dated May 2006 (ADAMS Accession Number Non-Proprietary version ML061560412)

15. Westinghouse letter to PNP, LTR-CDME-06-80-P, *Palisades Tubesheet Inspection Depth*, Revision 1 dated May 2006 (ADAMS Accession Number Non-Proprietary version ML061560408)

16. Westinghouse letter to PNP, LTR-CDME-07-22-P, *Responses to NRC Requests for Additional Information Regarding the Application of WCAP-16208-P, Revision 1, 'NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions' to the Palisades Nuclear Power Plant*, Revision 1 dated February 2007 (ADAMS Accession Number Non-Proprietary version ML070640062)

17. Entergy Nuclear Operations, Inc., letter to NRC, PNP 2011-11, *License Amendment Request for Steam Generator Cold-Leg Tubesheet Inspection*, dated March 3, 2011 (ADAMS Accession Number ML110680342)

18. NRC letter to Southern Nuclear Operating Company, Inc., *Vogtle Electric Generating Plant, Units 1 and 2, Issuance of Amendments Regarding Steam Generator Tube Inspection Alternate Inspection Criteria (TAC Nos. ME5067 and ME5068)*, dated March 14, 2011 (ADAMS Accession No. ML110660264).

19. Entergy Nuclear Operations, Inc., letter to NRC, PNP 2011-035, *Withdrawal of License Amendment Request for Steam Generator Cold-Leg Tubesheet Inspection*, dated April 19, 2011 (ADAMS Accession Number ML111090424)

20. NRC letter to Florida Power and Light Company, *Turkey Point Nuclear Generating Station Unit Nos. 3 and 4 – Issuance of Amendments Regarding Permanent Alternate Repair Criteria for Steam Generator Tubes*, dated November 5, 2012 (ADAMS Accession Number ML12292A342)

ATTACHMENT 2

Proposed Changes to

Palisades Plant
Renewed Facility Operating License DPR-20

and

Appendix A
Technical Specifications

Pages

(showing proposed changes; additions are highlighted and deletions are strikethrough)

14 pages follow

(1) Pursuant to Section 104b of the Act, as amended, and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," (a) ENP to possess and use, and (b) ENO to possess, use and operate, the facility as a utilization facility at the designated location in Van Buren County, Michigan, in accordance with the procedures and limitation set forth in this license;

(2) ENO, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) ENO is authorized to operate the facility at steady-state reactor core power levels not in excess of 2565.4 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~256~~XXX, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Fire Protection</u>

ENO shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the license amendment request dated December 12, 2012, as supplemented by letters dated February 21, 2013, September 30, 2013, October 24, 2013, December 2, 2013, April 2, 2014, May 7, 2014, June 17, 2014,

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

b. Performance criteria for SG tube integrity. (continued)

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

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "PCS 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 alternative repair criteria may be applied as an alternate to the 40% depth based criteria:

1. Tubes found by inservice inspection to contain flaws within 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service. |

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

c. Provisions for SG tube repair criteria. (continued)

2. Tubes found by inservice inspection to contain flaws within 12.5 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, shall be plugged. Flaws located below this elevation may remain in service.

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~~ and d.4~~, d.4 and d.5 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 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

4. When the alternate repair criteria of TS 5.5.8~~.~~c.1 are implemented, inspect 100% of the inservice tubes to the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8~~.~~c.1 every 24 effective full-power months, or one refueling outage, whichever is less.

5. When the alternate repair criteria of TS 5.5.8.c.2 are implemented, inspect 100% of the inservice tubes to the cold-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8.c.2 every 24 effective full-power months, or one refueling outage, whichever is less.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

d. Procedures for the recording and management of data,

e. Procedures defining corrective actions for all off-control point chemistry conditions, and

f. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.5.10 Ventilation Filter Testing Program

A program shall be established to implement the following required testing of Control Room Ventilation (CRV) and Fuel Handling Area Ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2 (RG 1.52), and in accordance with RG 1.52 and ASME N510-1989, at the system flowrates and tolerances specified below*:

a. Demonstrate for each of the ventilation systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the Fuel Handling Area Ventilation System when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
V-8A or V-8B	7300 ± 20%
V-8A and V-8B	10,000 ± 20%
V-95 or V-96	12,500 ± 10%

5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (continued)

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the Fuel Handling Area Ventilation System when tested in accordance with RG 1.52 and ASME N510-1989.

Ventilation System	Flowrate (CFM)
V-8A and V-8B	10,000 ± 20%
V-26A and V-26B	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
VF-66	6.00%	95%
VFC-26A and VFC-26B	0.157%	70%

d. For each of the ventilation systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_20)	Flowrate (CFM)
V-8A and V-8B	6.0	10,000 ± 20%
VF-26A and VF-26B	8.0	3200 +10% -5%

e. Demonstrate that the heaters for the CRV system dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
VHX-26A and VHX-26B	15 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

5.5 Programs and Manuals

5.5.11 Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

    1. API gravity or an absolute specific gravity,

    2. Kinematic viscosity, and

    3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program.

5.5.12 Technical Specifications (TS) Bases Control Program

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

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

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

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

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

5.5 Programs and Manuals

5.5.12 Technical Specifications (TS) Bases Control Program (continued)

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

d. Proposed changes that meet the criteria of Specification 5.5.12.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.13 Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

5.5.13 Safety Functions Determination Program (SFDP) (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leak Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, with the following exceptions:

1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at ≥10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 54.2 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

5.5 Programs and Manuals

5.5.14 Containment Leak Rate Testing Program (continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage is $\leq 1.0\ L_a$ when tested at $\geq P_a$ and combined with all penetrations and valves subjected to Type B and C tests. However, during the first unit startup following testing performed in accordance with this program, the leakage rate acceptance criteria is $< 0.6\ L_a$ when combined with all penetrations and valves subjected to Type B and C tests.

b) For each Personnel Air Lock door, leakage is $\leq 0.023\ L_a$ when pressurized to ≥ 10 psig.

c) For each Emergency Escape Air Lock door, a seal contact check , consisting of a verification of continuous contact between the seals and the sealing surfaces, is acceptable.

e. "Containment OPERABILITY" is equivalent to "Containment Integrity" for the purposes of the testing requirements.

f. The provisions of SR 3.0.3 are applicable to the Containment Leak Rate Testing Program requirements.

g. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

5.5 Programs and Manuals

5.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program. This documentation shall contain:

a) Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant superintendent.

5.5.16 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 Ventilation (CRV) Filtration, 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 whole body or its equivalent to any part of the body 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.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRV Filtration, operating at the flow rate required by the Ventilation Filter Testing Program, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

~~5.5 Programs and Manuals~~

~~This page retained for page numbering~~

5.6 Reporting Requirements

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

5.6.7 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

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

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

ATTACHMENT 3

Page Change Instructions and Revised Pages for

Palisades Plant
Renewed Facility Operating License DPR-20

and

Appendix A
Technical Specifications

14 pages follow

Page Change Instructions

ATTACHMENT TO LICENSE AMENDMENT NO. 2XX

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Remove the following pages of the Renewed Facility Operating License and Appendix A Technical Specifications and replace them with the attached revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

Renewed Facility Operating License

<u>REMOVE</u>	<u>INSERT</u>
Page 3	Page 3

Appendix A, Technical Specifications

<u>REMOVE</u>	<u>INSERT</u>
Section 5.5 Programs and Manuals Pages 5.0-12 through 5.0-22	Section 5.5 Programs and Manuals Pages 5.0-12 through 5.0-22
Section 5.5 Programs Manuals Page 5.0-28	Section 5.5 Programs Manuals Page 5.0-28

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(3) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) ENO, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

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(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. XXX, and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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5.5 Programs and Manuals

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2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gpm.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "PCS 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 alternative repair criteria may be applied as an alternate to the 40% depth based criteria:

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5.5.8 Steam Generator (SG) Program

c. Provisions for SG tube repair criteria. (continued)

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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 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

4. When the alternate repair criteria of TS 5.5.8c.1 are implemented, inspect 100% of the inservice tubes to the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8c.1 every 24 effective full-power months, or one refueling outage, whichever is less.

5. When the alternate repair criteria of TS 5.5.8c.2 are implemented, inspect 100% of the inservice tubes to the cold-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 5.5.8c.2 every 24 effective full-power months, or one refueling outage, whichever is less.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

5.5.9 Secondary Water Chemistry Program

A program shall be established, implemented and maintained for monitoring of secondary water chemistry to inhibit steam generator tube degradation and shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables,

b. Identification of the procedures used to measure the values of the critical variables,

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

d. Procedures for the recording and management of data,

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f. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

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a. Demonstrate for each of the ventilation systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the Fuel Handling Area Ventilation System when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Flowrate (CFM)
V-8A or V-8B	7300 ± 20%
V-8A and V-8B	10,000 ± 20%
V-95 or V-96	12,500 ± 10%

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5.5 Programs and Manuals

5.5.10 Ventilation Filter Testing Program (continued)

b. Demonstrate for each of the ventilation systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% for the CRV and < 1.00% for the Fuel Handling Area Ventilation System when tested in accordance with RG 1.52 and ASME N510-1989.

Ventilation System	Flowrate (CFM)
V-8A and V-8B	10,000 ± 20%
V-26A and V-26B	3200 +10% -5%

c. Demonstrate for each of the ventilation systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in RG 1.52 shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and equal to the relative humidity specified as follows:

Ventilation System	Penetration	Relative Humidity
VF-66	6.00%	95%
VFC-26A and VFC-26B	0.157%	70%

d. For each of the ventilation systems, demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is less than the value specified below when tested in accordance with RG 1.52 and ASME N510-1989:

Ventilation System	Delta P (In H_20)	Flowrate (CFM)
V-8A and V-8B	6.0	10,000 ± 20%
VF-26A and VF-26B	8.0	3200 +10% -5%

e. Demonstrate that the heaters for the CRV system dissipates the following specified value ± 20% when tested in accordance with ASME N510-1989:

Ventilation System	Wattage
VHX-26A and VHX-26B	15 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Ventilation Filter Testing Program frequencies.

* Should the 720-hour limitation on charcoal adsorber operation occur during a plant operation requiring the use of the charcoal adsorber - such as refueling - testing may be delayed until the completion of the plant operation or up to 1,500 hours of filter operation; whichever occurs first.

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5.5 Programs and Manuals

5.5.11 Fuel Oil Testing Program

A fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling requirements, testing requirements, and acceptance criteria, based on the diesel manufacturer's specifications and applicable ASTM Standards. The program shall establish the following:

a. Acceptability of new fuel oil prior to addition to the Fuel Oil Storage Tank, and acceptability of fuel oil stored in the Fuel Oil Storage Tank, by determining that the fuel oil has the following properties within limits:

    1. API gravity or an absolute specific gravity,

    2. Kinematic viscosity, and

    3. Water and sediment content.

b. Other properties of fuel oil stored in the Fuel Oil Storage Tank, specified by the diesel manufacturers or specified for grade 2D fuel oil in ASTM D 975, are within limits.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Fuel Oil Testing Program.

5.5.12 Technical Specifications (TS) Bases Control Program

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

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

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

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

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

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5.5 Programs and Manuals

5.5.12 Technical Specifications (TS) Bases Control Program (continued)

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

d. Proposed changes that meet the criteria of Specification 5.5.12.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.13 Safety Functions Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

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5.5 Programs and Manuals

5.5.13 Safety Functions Determination Program (SFDP) (continued)

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leak Rate Testing Program

a. A program shall establish the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated October 2008, with the following exceptions:

1. Leakage rate testing is not necessary after opening the Emergency Escape Air Lock doors for post-test restoration or post-test adjustment of the air lock door seals. However, a seal contact check shall be performed instead.

   Emergency Escape Airlock door opening, solely for the purpose of strongback removal and performance of the seal contact check, does not necessitate additional pressure testing.

2. Leakage rate testing at P_a is not necessary after adjustment of the Personnel Air Lock door seals. However, a between-the-seals test shall be performed at ≥10 psig instead.

3. Leakage rate testing frequency for the Containment 4 inch purge exhaust valves, the 8 inch purge exhaust valves, and the 12 inch air room supply valves may be extended up to 60 months based on component performance.

b. The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is 54.2 psig. The containment design pressure is 55 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

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5.5.14 Containment Leak Rate Testing Program (continued)

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage is $\leq 1.0\ L_a$ when tested at $\geq P_a$ and combined with all penetrations and valves subjected to Type B and C tests. However, during the first unit startup following testing performed in accordance with this program, the leakage rate acceptance criteria is $< 0.6\ L_a$ when combined with all penetrations and valves subjected to Type B and C tests.

b) For each Personnel Air Lock door, leakage is $\leq 0.023\ L_a$ when pressurized to $\geq$ 10 psig.

c) For each Emergency Escape Air Lock door, a seal contact check , consisting of a verification of continuous contact between the seals and the sealing surfaces, is acceptable.

e. “Containment OPERABILITY” is equivalent to "Containment Integrity" for the purposes of the testing requirements.

f. The provisions of SR 3.0.3 <u>are</u> applicable to the Containment Leak Rate Testing Program requirements.

g. Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

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5.5 Programs and Manuals

5.5.15 Process Control Program

a. The Process Control Program shall contain the current formula, sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR 20, 10 CFR 71, Federal and State regulations, and other requirements governing the disposal of the radioactive waste.

b. Changes to the Process Control Program:

1. Shall be documented and records of reviews performed shall be retained as required by the Quality Program. This documentation shall contain:

    a) Sufficient information to support the change together with the appropriate analyses or evaluation justifying the change(s) and

    b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after approval by the plant superintendent.

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5.5.16 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 Ventilation (CRV) Filtration, 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 whole body or its equivalent to any part of the body 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.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CRV Filtration, operating at the flow rate required by the Ventilation Filter Testing Program, at a Frequency of 18 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 18 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

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5.6 Reporting Requirements

5.6.6 Post Accident Monitoring Report

When a report is required by LCO 3.3.7, "Post Accident Monitoring Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to OPERABLE status.

5.6.7 Containment Structural Integrity Surveillance Report

Reports shall be submitted to the NRC covering Prestressing, Anchorage, and Dome Delamination tests within 90 days after completion of the tests.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

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

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

ATTACHMENT 4

Westinghouse Proprietary Class 2

SG-SGMP-10-4-P, Revision 1

Palisades Cold Leg Tubesheet Inspection Depth, C*

February 2010

(Proprietary Version)

<u>PROPRIETARY</u>

This attachment contains confidential information submitted under 10 CFR 2.390. Withhold from public disclosure.

16 pages follow

ATTACHMENT 5

Westinghouse Non-Proprietary Class 3

SG-SGMP-10-4-NP, Revision 1

Palisades Cold Leg Tubesheet Inspection Depth, C*

February 2010

(Non-Proprietary Version)

16 pages follow

Westinghouse Non-Proprietary Class 3

SG-SGMP-10-4-NP
Revision 1

February 2010

Palisades Cold Leg Tubesheet Inspection Depth, C*

Prepared for
Entergy Nuclear Operations, Inc.

Westinghouse

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SG-SGMP-10-4-NP
Revision 1

Palisades Cold leg Tubesheet Inspection Depth, C*

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EXECUTIVE SUMMARY

Nondestructive Examination NDE inspection by a qualified nondestructive examination technique to a defined inspection length below the top of the tubesheet ensures that steam generator tube burst and leakage requirements are met in the tubesheet region. A hot leg side NDE inspection length was provided in the C* generic topical report for Combustion Engineering designed steam generators (WCAP-16208, Revision 1, Reference 1) and supplemented for Palisades by Reference 3. This supplement provides the cold leg side inspection length in the event that a cold leg side examination is performed. The cold leg inspection lengths provided in this supplement have been developed using the methods and test data used in the C* generic topical report and the subsequent responses to the Nuclear Regulatory Commission (NRC) requests for additional information.

The cold leg inspection length is essentially the same as the hot leg inspection length reported in References 3 and 7. The resolution of the NRC equests for Additional Information (RAIs) on the submittal for the hot leg C*, Reference 7, were included in the calculations for the cold leg C* value. The inspection lengths in the table below provide assurance that the NEI 97-06 requirements for tube burst and leakage are met and that the conservatively derived maximum combined leakage from both tubesheet joints (hot and cold legs) is less than 0.2 gpm at accident conditions. This combined leakage of 0.2 gpm in the faulted loop is below the Palisades Nuclear Plant Technical Specification allowable accident-induced leakage of 0.3 gpm per steam generator.

Plant	Leak Rate Based Inspection Length Corrected for Dilation and NDE (inches)
Palisades Hot Leg	12.5
Palisades Cold Leg	12.5

1.0 INTRODUCTION

1.1 BACKGROUND

The PWR Owners Group program to provide recommended tubesheet (TS) region inspection lengths, for plants with Combustion Engineering supplied steam generators with explosive expansions, was documented in report WCAP-16208-P, Revision 1,Reference 1. This inspection length is commonly referred to as C* ("C-Star"). Reference 1 was first submitted to the NRC by other participants within the PWR Owners Group program. In preparation to submit Reference 1 to support the application of C* to Palisades, responses to NRC Requests for Additional Information (RAIs) which were relevant to the Palisades application were submitted to the NRC in Reference 2. A letter summarized the plant-specific application of Reference 1 to the hot leg side of the Palisades tubesheet, Reference 3 which incorporated the RAI responses.

The calculation of the C* inspection depth for the cold leg of the tubesheet at Palisades is similar to the calculation of C* for the hot leg. The only difference is the lower temperatures at the cold leg tube, tubesheet, and channelhead compared to the hot leg. The purpose of this document is to calculate the C* inspection depth for the cold leg side of the tubesheet using the same methods and techniques that were used to compute the C* inspection depth on the hot leg side of the tubesheet.

1.2 SUMMARY

The cold leg C* inspection distance has been calculated to include all of the effects that were included in the hot leg C* final documentation. The resulting cold leg C* distance is 12.5 inches below the bottom of the tube-to-tubesheet expansion transition. This value applies to each tube at the cold leg tubesheet region for the Palisades steam generators.

Leakage Based Inspection Length Including Tubesheet Deflection and NDE Corrections for Palisades Cold Leg

Plant	Leak Rate Based Inspection Length Adjusted for TS Dilation (inches)	Leak Rate Based Inspection Length Adjusted for TS Dilation and NDE (inches)
Palisades Cold Leg	[ ]^{a,c,e}	12.5

1.3 QUALITY ASSURANCE

The work that is presented in this document was completed and reviewed under the requirements of the Westinghouse Level II Policies and Procedures (Reference 4).

2.0 COLD LEG JOINT INSPECTION DISTANCE TECHNICAL APPROACH

The technical approach is the method used in the generic C* topical report (Reference 1) and the subsequent responses to the NRC requests for additional information (References 3 and 7).

2.1 ACCEPTANCE CRITERION

Acceptable joint length as reported for the hot leg joints in the C* topical report was determined by testing for two categories of concern: pullout load and leak rate. Pullout load and leak rate testing data were used to show compliance with the acceptance criteria (Reference 8). As reported in the C* topical report, the length needed to ensure that both the burst integrity and the leakage criteria are met was dominated in all cases by the threshold length required to meet the leakage criterion. Therefore, the leakage criterion defines the required cold leg tube-to-tubesheet joint length and bounds the inspection length for the cold leg side pullout criterion.

The C* generically applied limiting conditions for the leak rate criterion were based on a conservative assessment of conditions during a main steam line break (MSLB) event. Leak rate data in the C* analysis was evaluated at a pressure of 2560 psid and 600°F for the development of the hot leg inspection length. The pressure value of 2560 psid corresponds to the pressurizer safety valve setpoint plus 3 percent for valve accumulation less atmospheric pressure in the secondary side of the faulted steam generator. This pressure differential represents the pressure that would be obtained during a main steam line break due to total depressurization of the faulted steam generator with reactor coolant pressure rising to the setpoint of the reactor coolant system safety valves assuming no operator action to modulate or terminate safety injection. This pressure differential represents the limiting pressure that would create the most limiting leak rate.

As in the C* development for the hot leg side of the tubesheet, the accident-induced leak rate criterion for the Palisades Nuclear Plant is the plant-specific allowable value of 0.3 gpm per steam generator. In the C* generic topical work, Reference 1, the criterion was conservatively limited to 0.1 gpm per steam generator for this single type of flaw (tubesheet region cracking) representing all hot leg joints. The plant-specific threshold length for leakage is determined from the single joint leak rate as a function of the postulated flaw depth below the bottom of the expansion transition. The single joint leak rate must be less than or equal to the leak rate criterion of 0.1 gpm divided by the number of tubes assumed to be defective. The hot leg C* work determined an inspection length for the hot leg joints based on the assumption that all 7846 hot leg joints were leaking at the leak rate derived from the C* testing that would cumulatively equal 0.1 gpm. The allowable leak rate on this basis is 1.27E-05 gpm per hot leg joint. Note that the active tube count of 7846 tubes in the limiting SG is based on the active tube count at the time of the hot leg C* submittal. The current limiting active tube count is 7826 tubes in Steam Generator B.

The leak rate criterion for the sum of the cold leg joints and the hot leg joints if all are assumed to be leaking based on the method and reference transient used in Reference 1 is 0.2 gpm or two times the 0.1 gpm used in Reference 1. This criterion for leakage retains margin against the Palisades Nuclear Plant accident-induced leakage limit of 0.3 gpm/steam generator.

The following constraints guided the analysis for the development of the cold leg inspection length:

1. The acceptance criterion is the NDE inspection length in the cold leg tube-to-tubesheet joint that meets a total cold leg joint leak rate of 0.1 gpm per steam generator for the generic (Reference 1) MSLB case.

   Therefore, the total leak rate is 0.2 gpm per steam generator in the faulted loop which is two times the Reference 1 leak rate criterion of 0.1 gpm based on doubling the number of leaking joints by adding the cold leg joints to the hot leg joint count in the effected steam generator.

2. The inspection length must include consideration of the effect on leakage from:
   - Reactor coolant system (RCS) pressure and temperature adjustments to the leak rate test data,
   - Tube-to-tubesheet joint contact force adjustment resulting from the internal pressure and the RCS temperature, and
   - The tubesheet hole dilation caused by tubesheet deflection under primary-to-secondary pressure differential.

3.0 CALCULATION METHODOLOGY

3.1 BACKGROUND

Reference 1 provided the general methodology to determine the joint length that meets the leakage criteria. The applicable sections from this reference are as follows:

- Section 4.6 of Reference 1 describes how temperature affects the leak rate.
- Section 4.8 of Reference 1 describes how the leak rate data is evaluated to provide the joint length at which the leak rate criteria are met (prior to adjustments for NDE error and tubesheet hole dilation).

LTR-CDME-06-80-P, Revision 1 (Reference 3) and LTR-CDME-07-22-P (Reference 7) provide revised joint lengths for the hot leg under a "first slip" pullout load criteria and a lower temperature (583°F applicable to Palisades hot leg tubesheet joints). A similar methodology is employed in this report. The applicable inputs from these references are as follows:

- Figure 2 of Reference 7 provides the 95% lower bound prediction of first slip pullout force for 42 mil wall smooth bore tests. This figure is taken directly from the RAI response to RAI # 6 in Reference 7.
- Table 6-11 of Reference 1 provides the dilation axial force due to tubesheet bending. The bending on the cold leg side of the tubesheet is conservatively considered to be the same as the hot leg side of the tubesheet.
- The required engagement length of less than 5.25 inches to resist the three times the normal operation differential pressure (3NODP) pullout load of [ ]^{a,c,e} lb_f from Reference 3 is conservative for the cold leg because the value of "RCS Pressure and Diff Thermal Axial Force" of [ ]^{a,c,e} lb_f used in the calculation in Reference 3 is less than the value [ ]^{a,c,e} lb_f calculated at 532°F in Section 3.4. This is consistent with the discussion in the response to RAI # 5 in Reference 7.

3.2 TEMPERATURE CORRECTION FOR UNDILATED JOINT LENGTH

The effect of temperature on the leak rate from a tubesheet joint without tubesheet hole dilation was experimentally quantified in Reference 1. The effect of temperature on tubesheet hole dilation is accounted for analytically.

Section 4.6 of Reference 1 provides the experimentally determined relationship that describes how temperature affects the leak rate. This equation is used to adjust the leak rate data in Table 4-7 of Reference 1. The analysis that is described in Section 4.8 of Reference 1 is performed using the temperature-adjusted data to obtain the joint length that would meet the leakage criteria for an undilated tubesheet hole.

Reference 1 used a generic hot leg temperature of 600°F to determine the leakage-limited inspection distance. The Palisades cold leg temperature is 532°F (Reference 5). Section 2.3 of

Reference 3 demonstrated how the leak rate adjustment was applied for a hot leg temperature of 583°F applicable to Palisades. When a similar adjustment is made for a cold leg temperature of 532°F, then [

][a,c,e]

Table 1: WCAP-16208-P, Table 4-7: Transformed Leak Rate Data: Revised for Change of Temperature from 600°F to 532°F

Index	Sample	Temperature Corrected Data		Transformed	
		L Joint Length (inches)	**Q** Leak Rate at 532°F (gpm)	**$L-L_{avg}$** Joint Length (inches)	**$Q-Q_{avg}$** Leak Rate (gpm)

a,c,e

Calculations of the "Uncorrected Joint Length that Meets Leakage Criteria," then follow the methodology provided in Section 4.8 of Reference 1. The result is a minor revision to Figure 4-4 of Reference 1, as is shown in Figure 1. Using the leakage criteria as the y-axis leak rate value and reading the corresponding joint length off of the 95% confidence interval curve yields the "Uncorrected Joint Length that Meets Leakage Criteria."

a,b,c

Figure 1: WCAP-16208-P, Figure 4-4: Leak Rate vs. Joint Length at 532°F, ΔP=SLB, (Revised)

The result is that the "Uncorrected Joint Length that Meets Leakage Criteria," that was provided in Tables 4-9 and 6-15 of Reference 1 (at the assumed leak criterion of 0.1 gpm/SG), becomes [ ][a,c,e] inches when using the cold leg temperature of 532°F and the number of actual in-service tubes (7826 tubes/SG[1] – Reference 6).

3.3 FIRST SLIP PULLOUT FORCE

The first slip pullout force values are taken from Figure 2 of Reference 7. This figure shows the 95% lower prediction bound for the first slip pullout force as a function of joint length. The

1 The number of tubes considered here is the largest number of tubes in operation in either SG after the 2009 inspection.

"Axial Force" column of Table 2 contains these values of axial force from Table 1 (Column 2) of Reference 7.

3.4 RCS PRESSURE AND DIFFERENTIAL TEMPERATURE AXIAL FORCE

The impact on flow resistance of the tubesheet hole at the cold leg temperature is computed in a similar manner as was done in Reference 3. This calculation considers the temperature and the corresponding temperature dependent material properties to compute the interface force between the tube and tubesheet. Table 6-11 of Reference 1 uses a conservative value of [

]^{a,c,e}

3.5 DILATATION AXIAL FORCE

Table 6-11 of Reference 1 provides the dilation axial force due to tubesheet bending. The original design reports assume symmetry of the tubesheet for both hot and cold leg sides. Since the bending is due to the differential pressure across the tubesheet and the primary pressure on the cold leg side is slightly smaller than the primary pressure on the hot leg side, the magnitude of the tubesheet deflection from differential pressure is likely to be slightly smaller on the cold leg. In the current evaluation, the bending of the cold leg side is conservatively considered to be the same as the hot leg side of the tubesheet.

3.6 CALCULATION OF INSPECTION DISTANCE

Incorporating the "first slip" 95% lower bound prediction of Reference 7 and the 532°F value for the "RCS Pressure and Diff Thermal Axial Force" into Table 6-11 of Reference 1 yields a mechanism to adjust the "Uncorrected Joint Length that Meets Leakage Criteria" for tubesheet hole dilation.

Table 2 presents a revision to Table 6-11 of Reference 1 that accounts for "first slip" pullout data and a 532°F cold leg as described in the previous sections. The first column in this table is the depth measured from the expansion transition near the top of the tubesheet. The second column is the axial force from the expansion described in Section 3.3. The third column represents the axial force resulting from the internal pressure in the tube and the differential thermal expansion between the tube and the tubesheet corresponding to the cold leg temperature as described in Section 3.4. The fourth column is the sum of Columns 2 and 3. The fifth column shows the dilation force resulting from the tubesheet deflection described in Section 3.5. The sixth column shows the algebraic sum of Columns 4 and 5. Since a negative axial force is not possible (when there is no radial contact force between the tube and the tubesheet), negative values are truncated to 0. The subsequent columns follow the computation as described in Reference 1.

Section 2.3 presented "Uncorrected Joint Length that Meets Leakage Criteria" length of [ ][a,c,e] inches for leakage criteria of 0.1 gpm/SG for 7826 tubes/SG. Looking up each "Uncorrected Joint Length that Meets Leakage Criteria" in the rightmost column of Table 2, and interpolating to find the result in the leftmost column of the table, produces "Joint Length that Meets Leakage Criteria" value of [ ][a,c,e] inches. Adding NDE axial position uncertainty of [ ][a,c,e] inch to both values yields a leakage-based inspection length of 12.50 inches. This combination of the computed C* value and the axial position uncertainty is consistent with the discussion of RAI #8 in Reference 7. Note that this length is measured from the bottom of the expansion transition, not the top of the tubesheet. This C* value for the cold leg of 12.5 inches is the same as the C* value for the hot leg.

3.7 COMPARISON WITH INSPECTION DISTANCE FOR HOT LEG

The C* inspection distance for the hot leg at Palisades Nuclear Plant was computed in References 3 and 7. Variations in the way the uncertainties were treated made only a slight difference in the results. One of the conservatisms in the computation of the hot leg C* was the use of the generic value of [ ][a,c,e] lb_f for RCS Pressure and Diff. Thermal Axial Force which was discussed in Reference 7. If the plant-specific value of [ ][a,c,e] lb_f is used in the computation, the value of hot leg C* would be reduced. The appropriate plant-specific value for the cold leg conditions [ ][a,c,e] lb_f is used for the calculation of the cold leg C*. The increase in the C* length due to the increased leak rate at the lower temperature coincidentally is within round-off of the increase in the hot leg C* due to the use of the conservative Axial Force value. Therefore the C* value of 12.5 inches for the cold leg is the same as the C* value for the hot leg because of the additional conservatism of the hot leg calculation.

Table 2: WCAP-16208-P, Table 6-11: Effect of Tubesheet Deflection for Palisades Steam Generators: Revised for Use of First Slip Loads and 532°F Cold Leg

Depth in Tubesheet (in)	Axial Force (lb_f)	RCS Pressure and Diff. Thermal Axial Force (lb_f)	Initial Axial Force (lb_f)	Dilation Axial Force (lb_f)	Net Axial Force (lb_f)	Net / Initial Ratio	Equiv. No-Dilate Length (inch)	Cum. No-Dilate Length (inch)

a,b,c

4.0 REFERENCES

1. Westinghouse Report WCAP-16208-P, Revision 1, "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions," May 2005.

2. Westinghouse Letter LTR-CDME-06-40-P, Revision 1, "Comments on the Application of WCAP-16208-P Revision 1, 'NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions' to the Palisades Nuclear Power Plant," May 2006.

3. Westinghouse Letter LTR-CDME-06-80, Revision 1, "Palisades Tubesheet Inspection Depth," May 2006.

4. "Westinghouse Level II Policies and Procedures," Rev. 0, Effective August 3, 2009.

5. Westinghouse Letter LTR-CDME-07-124, "Entergy Input to Palisades REFOUT 19 Degradation Assessment," June 8, 2007.

6. Westinghouse Report SG-SGMP-09-3, "Palisades Nuclear Plant 1R20 Outage Steam Generator Condition Monitoring Report," April 2009.

7. Westinghouse Letter LTR-CDME-07-22-P, "Responses to NRC Requests for Additional Information Regarding the Application of WCAP-16208-P, Revision 1, 'NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions' to the Palisades Nuclear Power Plant," February 2007.

8. NEI 97-06, Revision 1, "Steam Generator Program Guidelines," Nuclear Energy Institute, Washington, DC, January 2001.

ATTACHMENT 6

Westinghouse Electric Company

Affidavit
CAW-10-2752

Application for Withholding Proprietary Information from Public Disclosure

November 17, 2010

8 pages follow

Westinghouse

Westinghouse Electric Company
Nuclear Services
1000 Westinghouse Drive
Cranberry Township, PA 16066
USA

U.S. Nuclear Regulatory Commission
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Proj letter CPAL-10-39

CAW-10-2752

November 17, 2010

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: SG-SGMP-10-4-P, Revision 1, "Palisades Cold Leg Tubesheet Inspection Depth, C*" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-10-2752 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Entergy Nuclear Operations, Inc.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-10-2752 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, Suite 428, 1000 Westinghouse Drive, Cranberry Township, PA 16066.

Very truly yours,

BHMaurer /for

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

Enclosures

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

BFMaurer

B. F. Maurer, Manager
ABWR Licensing

Sworn to and subscribed before me
this 17th day of November 2010

Renee' Giampole
Notary Public

COMMONWEALTH OF PENNSYLVANIA
NOTARIAL SEAL
Renee Giampole, Notary Public
Penn Township, Westmoreland County
My Commission Expires September 25, 2013

(1) I am Manager, ABWR 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 Proprietary Information from Public Disclosure accompanying this Affidavit.

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

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

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

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system 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 SG-SGMP-10-4-P, Revision 1, "Palisades Cold Leg Tubesheet Inspection depth, C*" (Proprietary) dated February 2010, for submittal to the Commission, being transmitted by Entergy Nuclear Operations, Inc. letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with implementing a steam generator tube alternate repair criterion called C* that does not require an eddy current inspection and plugging of the tubes below a distance of 12.5 inches from the top of the tubesheet on the cold leg side of the tubesheet region and may be used only for that purpose. The cold leg inspection length is essentially the same as the hot leg inspection length that has been previously approved for Palisades.

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

(a) Provide documentation of the analyses, methods, and testing which support the implementation of an alternate repair criterion, designated as C*, for a portion of the tubes within the cold leg side of the tubesheet region of the Palisades steam generators.

(b) Assist the customer in obtaining NRC approval of the Technical Specification changes associated with the 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 purpose of meeting NRC requirements for licensing documentation.

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

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

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar 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).

COPYRIGHT NOTICE

ATTACHMENT 7

Westinghouse Proprietary Class 2

LTR-SGMP-15-88, Rev. 1 P-Attachment

Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis

February 23, 2016

(Proprietary Version)

20 pages follow

ATTACHMENT 8

Westinghouse Non-Proprietary Class 3

LTR-SGMP-15-88, Rev. 1 NP-Attachment

Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis

February 23, 2016

20 pages follow

Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis

February 23, 2016

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis

In 2011, Entergy submitted a license amendment request (LAR) for implementation of the C* alternate repair criteria (ARC) to the cold leg tube-in-tubesheet region of the Palisades steam generators (SGs). A request for additional information (RAI) from the Nuclear Regulatory Commission (NRC) dated April 5, 2011 was received asking to justify the adequacy of the proposed C* distance in light of lessons learned from the H* review (if applicable).

The RAI requested that supplemental information be provided by April 19, 2011. The requested information could not be provided by the requested date thus the amendment request was withdrawn. Subsequently, this review was performed and results provided herein. The subject areas judged to potentially impact the C* analyses are identified below.

a. Calculate how material property variances might affect the pullout length (full bundle versus deterministic methods).
b. Determine how divider plate cracking might affect the repair criteria.
c. Review how the potential for cracking might affect the cold leg C*.
d. Compare CE-type explosively expanded and Westinghouse-type hydraulically expanded tube-tubesheet joint leak rates.

This document addresses these subject areas. This review concludes that lessons learned from the H* program are not applicable to, nor do they negatively affect the C* ARC. The H* ARC is applicable to hydraulically expanded tube-in-tubesheet joints. The C* ARC is applicable to tube-in tubesheet expansions produced using the Combustion Engineering (CE) Explansion explosive expansion process. The Explansion process is similar to the WEXTEX explosive expansion process applied in Westinghouse steam generators. The ARC applicable to WEXTEX explosive expansions is termed W*. Explosively expanded joints have been shown to produce substantially more robust joint characteristics compared to hydraulic expansions. Evaluation methodologies developed under the W* program have been utilized herein to emphasize differences between H* and C* and to identify areas of conservatism under the C* program.

For this evaluation a conservative cold leg temperature of 532°F consistent with References 1 and 4 was used; the actual normal operating cold leg temperature is 537°F. The cold leg temperature of 532°F represents the SG temperature at hot zero percent power condition combined with steam line break (SLB) assumptions.

Detailed responses addressing the above subject areas are provided below:

a. The C* tubesheet deflection model is a two-dimensional (2-D) axisymmetric model and considered conservative. A full channelhead complex (tubesheet, channelhead, stay cylinder, and lower stub barrel) three-dimensional (3-D) finite element model is not considered necessary. This is due to the axisymmetric nature of the C-E SG design, which includes a central stay cylinder that functions to limit tubesheet deflection due to bending during operation and under accident conditions. Tubesheet rotations are expected to be uniform along specific radial locations about the central axis of the SG. Deflections are dictated by the radial distance from the centerline of the SG.

The limiting radial location about the SG centerline with regard to calculated deflections is used in the C* analysis. In the C-E design the divider is bolted in place and does not influence tubesheet rotations.

Previous H* efforts have shown that the application of a fully probabilistic model can reduce the defined H* value compared to the original analyses. Both 2-D and 3-D models were developed for the Westinghouse Model 51F SG. Evaluation of tubesheet deflection for the Model 51F SG shows that 2-D model deflections bound the 3-D model results. As such, the existing C* analysis for Palisades is judged acceptable and expected to produce a conservative result.

The C* criteria is comprised of two components, each with a specified distance of non-degraded tube which must be established to ensure the basis is satisfied; a structural, or pullout component, and a leakage component. The structural component develops a non-degraded expanded tube length which satisfies the pullout criterion. This value is 5.25 inches for Palisades, and is based on the first slip pullout data (Reference 4). The pullout resistive capability is determined by a number of inputs; hole dilation, residual contact force due to the expansion process, and contact loads developed by internal tube pressurization and thermal expansion. Variances in the tube and tubesheet material property values could negatively affect the structural length value of 5.25 inches (Reference 1); however the impact is expected to be small. Since the criteria inspection depth is dictated by the leakage length (12.5 inches), variances in the structural length will be overshadowed by the leakage inspection length. The net effect to resistive load capability if tube and tubesheet coefficients of thermal expansion act in a direction adverse to the analysis is approximately [ ]^{a,c,e}. This value is approximately 1/5th of the nominal contribution due to pressure and thermal expansion. Thus, the pullout length would not be expected to be affected by more than about 1 inch.

The remainder of this response focuses on the impact of variability in the applied material property variances upon the C* (leakage) distance since the leakage distance is approximately 2.5 times the pullout distance.

<u>Material Property Variance:</u>

A substantial testing program was performed as part of the H* analysis that investigated the variance in tube and tubesheet material coefficient of thermal expansion. This program concluded that the standard deviation of the coefficient of thermal expansion for the tube is 2.33% and for the tubesheet, 1.44% (Reference 2).

Similar RAIs were submitted in 2009 to a utility regarding application of the W* alternate repair criteria in the cold leg of the SGs. The RAI response (Reference 5) considered the impact of a one standard deviation variance for both the tube and tubesheet. This approach was judged conservative by Westinghouse at the time and this judgment remains today since there is no apparent metallurgical theory to explain how this variance could be present.

A sensitivity study was performed to investigate the impact upon the C* limiting distance for variance in coefficient of thermal expansion. The nominal coefficient of thermal expansion (CTE) and Young's Modulus values were taken from the ASME Code 2001 through 2003 Addenda

(Reference 6) for the evaluation contained herein. The original cold leg C* value calculated for Palisades in 2010 was determined to be 12.50 inches (including NDE measurement uncertainty on the crack elevation below TTS) and is consistent with the hot leg C* value. Using the 2001 through 2003 Addenda CTE values, the nominal cold leg C* dimension (including NDE measurement uncertainty on the crack elevation below TTS) is determined to be 12.53 inches for the hot leg and 12.79 inches for the cold leg. This study shows that the impact upon the C* value is small for CTE variances. The difference of 0.29 inches in the cold leg C* distance calculated herein are attributed to use of the 2001 through 2003 Addenda material properties.

Table 1 presents the mean cold leg C* value and the cold leg C* values for positive and negative variances of 1 standard deviation for the tube and tubesheet. An increase in the CTE value for the tubesheet (larger hole enlargement) is conservative, while a decrease in the CTE value for the tube (less tube diametrical expansion) is conservative. For a 1 standard deviation variance in the conservative direction for both tube and tubesheet, the cold leg C* leakage value is increased to 13.67 inches; an increase of 0.88 inches. Note that variance in the conservative direction for both tube and tubesheet is required for this reduction. The SRSS Combination is a square root of sum-of-squares of the tube only and tubesheet only CTE variances.

Table 1: Summary of Cold Leg C* Distances for Various CTE Conditions			
	SLB Conditions Cold Leg C* Distance (inches)	Variance from Mean C* (inch)	SRSS Combination (inches)
C* Mean Properties (1)	12.79	N/A	N/A
C* Tube CTE minus 1 SD	13.33	+0.54	N/A
C* Tube CTE plus 1 SD	12.35	-0.44	N/A
C* Tubesheet CTE minus 1 SD	12.53	-0.26	N/A
C* Tubesheet CTE plus 1 SD	13.10	+0.31	N/A
C* Tube –1 SD, Tubesheet +1 SD	13.67	+0.88	13.41

(1): Based on 2001 through 2003 Addenda

A Monte Carlo combination of the tube and tubesheet variances was performed to determine the frequency at which simultaneous combination of CTE variances in the conservative direction could occur. The probability of having this combination was determined to be 2.3%. Thus, 97.7% of the time one of the CTE variances will be bounded by a 1 standard deviation variance in the conservative direction.

Any postulated C* leakage length associated with CTE variances is likely to be inspected given that it is standard practice to "over collect" rotating pancake coil (RPC) probe data in the tubesheet region. This is done to avoid retesting tubes with insufficient data collection lengths. Current practice at Palisades is to collect RPC data to 1 inch below the C* distance (Reference 7).

In summary, the potential effects of tube and tubesheet material CTE variances have a negligible impact upon the calculated C* distance given that RPC inspection is performed to a depth one inch greater than the C* distance, the probability of both tube and tubesheet CTE simultaneously existing in a direction negative to the C* distance is low, and the C* distance is calculated using the limiting radial location based on tubesheet deflection. Finally, given that the C* distance is calculated using the limiting radial location, which applies only to a small percentage of all tubes, combining this percentage with the probability of both tube and tubesheet CTE acting in a negative fashion, the ultimate combined probability that for any particular tube the true C* distance would not be described by the proposed 12.5 inches distance is less than 2.3%.

b. The observation of divider plate cracking in foreign units will not affect the Palisades C* distance. The C-E SG design uses a bolted divider plate. A "divider bar" is welded to the tubesheet, channelhead, and the vertically oriented surface of the stay cylinder. The divider plate is then bolted to the divider bar. Thus, in the C-E SG design, the divider plate cracking reporting in foreign units is not applicable and therefore has no impact upon the C* criteria. Entergy has performed visual inspections of the divider bar welds; no degradation has been reported.

c. An underlying premise of the C* methodology is that the leakage limiting distance is based on the assumption that all tubes contain a complete circumferential separation just below the C* leakage distance. A SLB conditions leakage contribution limit of 0.1 gpm per leg was applied. The undilated leakage length is the length with associated leak rate such that when the number of tubes in the SG is multiplied by the leak rate, the total leak rate for the leg is 0.1 gpm. This assumption is exceptionally conservative. The following discussion is provided to support this conclusion, and the best estimate of leakage, up to the end of the current operating license, is essentially zero from the cold legs.

<u>Initiation Potential:</u>

Since indications detected within the C* distance must be plugged, the only potential leakage source is from postulated indications below the inspection distance.

Under the W* ARC, potential leakage from indications below the W* distance were estimated based on the elevation trending of historic hot leg indications. Figure 1 presents a histogram plot of the hot leg PWSCC elevation (below top-of-tubesheet) distribution for Palisades and plant SQN-2, a W* plant. This plot shows the initiation susceptibility for Palisades is reduced compared to SQN-2. This reduction is likely attributed to the reduced T_{hot} value of Palisades as residual stresses associated with the WEXTEX and Explansion explosive expansion processes are expected to be similar. Figure 1 shows the SQN-2 PWSCC initiation is more concentrated in the expansion transition region, 0 to 1.0 inch below the top-of-tubesheet whereas the Palisades elevation distribution is more evenly distributed over the inspected tube-in-tubesheet length. The tubesheet drilling processes used for both plants may also influence this potential. The SQN-2 tubesheet tube holes were gun drilled. Gun drilling has a greater potential for localized tubesheet drilling anomalies such as gouges in the tubesheet, particularly nearer the top of tubesheet, which then are manifested as bulges or overexpansions when the tube is expanded into the tubesheet. The bore trepanning (BTA) process applied for production of the tubesheet tube holes at Palisades reduces the potential for tubesheet tube hole anomalies.

Previous evaluations of residual stress condition (Reference 3) within expanded tube-in-tubesheet conditions have determined that residual stresses in the fully expanded region are compressive on the ID, thus the likelihood of PWSCC, with the exception of localized bulge or overexpansion conditions, is limited. The Palisades tubesheet tube holes were formed using a bore trepanning (BTA) process, thus the likelihood of localized bulges and overexpansions is reduced. Previous evaluations of eddy current noise conditions within the Palisades expanded tube-in-tubesheet region have shown the Palisades noise condition to be reduced compared to other C-E SGs which used a gun drilling process. This supports the judgment that the BTA process results in a lesser potential for localized gouges in the tubesheet tube hole, and thus, a reduced potential for tube bulges and overexpansions resultant from tube expansion into the tubesheet hole anomaly.

Previous W* applications utilized a regression fit to the primary water stress corrosion cracking (PWSCC) elevation below top-of-tubesheet data to estimate the primary-to-secondary leak rate during accident conditions. A similar evaluation was performed for Palisades. The W* methodology divided the W* distance into two regions; from top-of-tubesheet (TTS) to 4 inches below and from 4 inches to 8 inches below TTS. A typical inspection distance of 8 inches below TTS was applied. The indications in the 8 to 12 inches below TTS region were then estimated from this binning. Indication leakage at elevations greater than 12 inches below TTS was taken as zero since the hole dilation will turn positive below the tubesheet neutral axis.

Figure 2 plots the elevation distribution of the Palisades PWSCC indications. If a similar methodology is applied, 10 indications are found in the upper half of the C* distance and 6 indications are found in the lower half of the C* distance. A 95% prediction bound to the indication data was developed in Figure 2. This plot shows that for data through the most recent inspection, the number of postulated hot leg indications below the C* distance for both SGs combined in the one inch bins from the C* to the tube end is two. Therefore, no more than about 16 indications ((20.5 inches (PNP tubesheet thickness) – 12.5 inches (PNP C* depth))*2 indications/inch) would be postulated for both SGs combined. If all 16 are assumed in one SG, the expected SLB leak rate is the ratio of 16 divided by the number of active tubes (7911) utilized in the original C* analysis, or a SLB conditions bounding leak rate of 2.02×10^{-4} gpm in any SG. The C* methodology associates the defined C* leakage distance with the assumption of all tubes containing a circumferential separation just below the C* elevation. Figure 2 shows that this assumption is exceptionally conservative as at the upper 95% prediction to the existing indication elevation data, 16 potentially flawed tubes are currently present below the C* distance on the hot leg side. Figure 3 also supports this judgment since at 40 effective full power years (EFPY) on the SGs, the estimated probability of failure, adjusted to a hot leg temperature of 600°F is approximately 30% of the tubes. For the current hot leg temperature of 583°F, at 40 EFPY the estimated probability of failure is approximately 4% of the tubes. Figure 3 shows the SQN-2 initiation data to be reduced for the last few inspections compared to prior inspections. This reduction in initiation coincides with the introduction of zinc injection into the primary system. Palisades also utilizes zinc injection.

Figure 3 compares the tubesheet region PWSCC initiation characteristics of various plants with Alloy 600 mill annealed tubing, explosively expanded through the tubesheet thickness, which have been approved for either the W* or C* ARC. The Palisades data was adjusted to a 600°F basis using an Arrhenius equation with activation energy of 33 kcal/mole. Westinghouse traditionally has used an activation energy value of 33 kcal/mole for PWSCC initiation. Values ranging from 25 to

50 kcal/mole have been used in various industry evaluations. This data shows the Palisades initiation function to be consistent with other plants using explosive tube expansion for closure of the tube-to-tubesheet crevice. Note that the reduction of initiation potential for plant SQN-2 at approximately 11 EFPY is associated with the initiation of zinc injections into the primary system. Palisades also injects zinc into the primary system. Predictions based on the historic data should be conservative as zinc injection should reduce PWSCC initiation.

In order to estimate cold leg PWSCC initiation potential, a Weibull initiation potential analysis is performed for the hot leg. The characteristic life, or the point in time when 63.2% of the population is predicted to be affected is adjusted based on the temperature difference between the hot leg and cold leg. The temperature adjustment factor is developed using an Arrhenius equation. The Weibull slope is assumed to be constant for both hot and cold legs. Using the Weibull failure analysis equation, Weibull slope, and adjusted cold leg characteristic life, the number of postulated indications can be predicted for any time element. This methodology has been used in numerous evaluations.

For activation energy of 33 kcal/mole, this adjustment factor is 4.36 for a hot leg temperature of 583°F and cold leg temperature of 532°F. The characteristic life for the limiting SG with regard to PWSCC initiation potential, SGA, is 67 EFPY for the hot leg. The expected cold leg characteristic life is then the adjustment factor times the hot leg characteristic life, or 292 EFPY. The Weibull slope, cold leg characteristic life, and tube count are then used with the Weibull failure equation to estimate the number of indications postulated to be present on the cold leg at various EFPYs. The smallest observed hot leg to cold leg improvement factor for any plant which has reported cold leg cracking was 3.50. This adjustment factor was used instead of the value of 4.36, which then gives cold leg PWSCC characteristic life of 235 EFPY. For these inputs, the first cold leg PWSCC initiation is expected to occur by 29.5 EFPY (1R31 outage, 2026), two indications are predicted by 1R35 (2032), and 0.1% of the population is expected to be affected by 1R44 (48 EFPY), which is beyond the current operating license of 2031 or 1R35. It should be noted that application of a Weibull initiation model is considered conservative as an underlying assumption of the Weibull distribution is that the population evaluated is homogenous, i.e., all tubes have equal PWSCC initiation potential. Certainly this is not the case due to variation in individual heat microstructure, heat treatment (annealing), and influence from bulges and overexpansions.

As the time to achieve 0.1% cold leg tubes affected with PWSCC is beyond the operating license of the plant, any potential increase in the C* distance for the cold leg associated with variance in material properties will be overcome by the lack of indications that potentially could contribute to primary-to-secondary leakage during a SLB event.

Furthermore, below the neutral axis of the tubesheet, hole dilation due to bending becomes positive for the majority of the tubes. That is, the hole becomes smaller. This statement applies only to the hole itself. The analysis considers contributions to the composite axial force from the expansion process, pressure and thermal effects, and hole dilation. The elevation within the tubesheet when the composite force becomes positive is well above the tubesheet neutral axis. Thus, leakage potential is further reduced below the neutral axis. The contact force reduction due to dilation from the axisymmetric deflection model turns positive at [ ]^{a,c,e} inches below the top-of-tubesheet for the limiting radial location. Thus for the limiting radial location it is a reasonable judgment that indications at greater than [ ]a,c,e inches below the cold leg TTS represent a zero leakage potential. This discussion regarding hole dilation below the neutral axis shows there is additional conservatism inherently present in the C* ARC.

d. Leakage characteristics of explosively expanded and hydraulically expanded tube-in-tubesheet conditions are not compatible.

The leakage characteristics of hydraulically expanded tubing and explosively expanded tubing are not expected to be similar. Pull testing has shown explosively expanded tubing to exhibit significantly larger resistive load capability compared to hydraulically expanded tubing for equal tube-to-tubesheet joint lengths. The difference implies a substantially larger residual contact pressure between the tube and tubesheet which would then influence leakage characteristics.

Figure 4 presents a plot of the C* (smooth) and H* leakage data at room temperature conditions and pressure differential approximately equal to SLB conditions. Figure 4 contains two subsets of H* data; for hydraulic expansion pressures of 30 to 31 ksi and for >32 ksi expansion pressure. The >32 ksi expansion pressure data shows a slight reduction in leak rate compared to the 30 to 31 ksi expansion pressure data. The H* leak rates for a joint length of 5.5 inches range from [

]a,c,e.

Figure 5 presents a plot of the C* (smooth and rough bore) data, W* data, and H* data at elevated temperature conditions and pressure differential approximately equal to SLB conditions. This plot shows a general trending of the W* and C* leak rate data to reduce with increasing joint lengths. The H* data contains only 16.5 inch joint lengths; the observed H* leak rates are substantially [

]a,c,e.

Figure 6 shows only the W* and C* elevated temperature data with a reduced range of the x-axis data to show greater detail. This plot accentuates the observed trending of reduced leak rate with increasing joint length at elevated temperature conditions.

Figure 7 shows the regression of C* leakage data at elevated temperature and SLB conditions and includes an upper 95% confidence interval to the data. This plot implies that at the upper 95% prediction interval, the leak rate is zero at approximately [ ]a,c,e below the TTS. This figure is taken from Reference 4 and includes the "updated uncertainty analysis" method as suggested by the NRC. The same figure is applicable to the hot leg C* application.

<u>Comparison of Explansion and Hydraulically Expanded Tubing Leak Rates</u>

Room temperature leakage testing for the hydraulically expanded tubing indicates average leak rates of [ ]a,c,e gpm for a pressure differential of 1910 psi and [ ]a,c,e gpm for a pressure differential of 2650 psi with a crevice length of 16.5 inches. In comparison, the 3/4 inch smooth bore OD tube Explansion room temperature leak rate data indicates an average leak rate of [ ]a,c,e gpm at a pressure differential of 1900 psi and [ ]a,c,e gpm at a pressure differential of 2560 psi for crevice lengths ranging from 2 to 5.5 inches. Thus for similar pressure differential conditions the hydraulically expanded tube leak rates are about [ ]^{a,c,e} times greater with a crevice length which is about 4 times the crevice length of the C* Explansion specimens.

Elevated temperature testing shows a similar trending between the two designs. The average elevated temperature leak rate for the hydraulically expanded tubing specimens at 2650 psi differential pressure is [ ]^{a,c,e} gpm for a 16.5 inch crevice length whereas the average leak rate for the 3/4 inch smooth bore OD Explansion specimens is [ ]^{a,c,e} gpm for crevice lengths ranging from 2 to 5.5 inches. If the adjustments for circumference and crevice length are included, the adjusted Explansion leak rate (using the equation below on Page 10) for comparison purposes is [ ]^{a,c,e} gpm, or about [ ]^{a,c,e} times less than the hydraulically expanded tubes. The C* data is similar to the W* data in that the data shows a reduction in leak rate with increasing crevice length. For the smooth bore C* specimens with 4 and 5.5 inch crevice lengths, the average leak rate at elevated temperature and 2560 psi differential pressure is [ ]^{a,c,e} gpm, or about [ ]^{a,c,e} times less than the hydraulically expanded tubes, and when the adjustment for crevice length is performed for these tubes, the Explansion leak rates are [ ]^{a,c,e} times less than the hydraulically expanded tubes for equal crevice lengths.

Clearly there are systematic differences between the two expansion processes with regard to residual contact pressure due to the expansion process and with regard to effective contact pressure at elevated temperature conditions.

If the tube hole and tube are examined on a microscopic level, the tube hole surface is quite irregular due to the drilling process. By rule of thumb, the peak to valley dimension of a 250 RMS surface finish is 4 times the RMS value, or 0.001 inch. If the tube and tubesheet are in intimate contact at the peaks of the surface finish an average tube-to-tubesheet radial gap of 0.0005 inch is established. The theory of flow between infinite parallel plates can be used to trend leakage characteristics for varying radial gaps. For a cylindrical condition the equation for leakage flow in volume is;

$$Q = \frac{\Pi D a^3 \Delta P}{12 \mu L}$$

Where D = diameter of piston, a = crevice gap in radial direction, μ = absolute viscosity, L = axial length of crevice.

If a starting radial gap of 0.0005 inch (one half of 0.0010 inch, from above) and room temperature conditions are used, the calculated hydraulic expanded tube leak rate for a pressure differential of 1910 psi and 16.5 inch crevice length is $4.66\text{x}10^{-3}$ gpm, or slightly greater than the average of the room temperature 1910 psi leak rate data of [ ]^{a,c,e} gpm. In order to achieve the average leak rate for the 1910 psi data the effective crevice gap must be *reduced* from the initial assumption of gap. If the pressure of 1910 psi is applied to the tube inside diameter (ID), tube outside diameter (OD), and tube hole, the calculated gap between the tube and tubesheet must then be *increased* (since equal pressure would act on the tube OD and ID creating a neutral tube expansion condition).

Thus, the application of the crevice pressure in a pure sense, i.e., the crevice pressure results in hole dilation assuming the pressure is uniformly applied to the tube hole, [ ]a,c,e.

This observation is even more pronounced for the Explansion tests. Using the room temperature, 4 inch crevice data with a starting crevice gap of 0.0005 inch, the calculated leak rate is 2.12×10^{-2} gpm at a 1910 psi pressure differential while the observed average leak rate is [ ]a,c,e gpm. In order to achieve a leak rate of [ ]a,c,e gpm, the 0.0005 inch starting crevice gap must be reduced to [ ]a,c,e inch. Thus while the hydraulically expanded leak rate data is closely approximated using the above equation, the Explansion test data shows the above equation grossly overestimates the leakage performance, suggesting that the explosively expanded condition is much more resistant to leakage than the hydraulically expanded data.

For the elevated temperature, 4 and 5.5 inch crevice C* specimens, using an average crevice length of 4.75 inches, the gap required to achieve the average leak rate is [ ]a,c,e inch. For the same specimens at room temperature, the gap required to achieve the average leak rate is [ ]a,c,e inch. The spreadsheets used to calculate the contact forces identifies a gap closure going from room temperature to the hot leg operating temperature of [ ]a,c,e inch, which matches well with the calculated difference between the gaps required to achieve the average leaks of [ ]a,c,e inch. Thus the information developed within the sum of forces approach can be used to identify the elevation in the tubesheet when the applied positive contributing forces (pressure, thermal, and process) exceed the reduction due to dilation. This elevation is approximately at the [ ]a,c,e.

For the limiting radial location on the tubesheet, the sum of forces for the one inch length below the C* distance was used to estimate the amount of crevice closure due to the combined loading condition. This evaluation suggests that the applied forces will result in tube deflection of approximately [ ]a,c,e radially, which slightly exceeds the tube-to-tubesheet gap associated with the elevated temperature leak test specimens. Thus, below the C* distance, [ ]a,c,e.

<u>Discussion of H* Crevice Pressure Tests</u>

Westinghouse studied the effect of crevice pressure by performing pressure tests on 11/16 inch OD hydraulically expanded tube-in-tubesheet collar specimens. In this testing, a 9.0 inch long hydraulically expanded tube length was used in a 14 inch long tubesheet simulant collar. The tube above the collar (secondary side) was weld plugged. Pressurized fluid was supplied to the tube by a welded pressure fitting. Drilled holes in the tube wall located below the hydraulically expanded length provided for direct fluid communication with the tube-to-tubesheet crevice. Pressure taps were located on the tubesheet collar at 7.5, 5.5, 4.5, 3.5 and 1.5 inches below the top of collar, or approximately 2, 4, 5, 6, and 8 inches above the bottom of the hydraulically expanded length.

These tests indicate pressure in the crevice was elevated compared to secondary side pressures and that the fluid remained in a liquid state up to an elevation near the secondary side elevation. These tests show that the pressure within the crevice was slightly reduced compared to the primary side pressure within the [ ]^{a,c,e}. Therefore, the assumption that the pressure acting on the tubesheet tube hole throughout the crevice length is equal to the primary side pressure is conservative.

Leakage data was measured however this data is not used within the H* criterion. This testing was performed for hydraulically expanded tubes. This condition is systematically different from an explosively expanded tube condition. In particular, the interface between the tube and tubesheet in the as-produced condition is dramatically different and is confirmed by a comparison of leak rate data for the two configurations.

<u>Summary of Leakage Potential Discussion:</u>

This discussion has concentrated on identifying the differences in C* and H* leak rates which supports judgment of a systematic performance difference between explosively expanded tubes and hydraulically expanded tubes. Therefore, potential lessons learned from the H* program are judged not relevant to the C* analysis. Additionally, methods developed under the W* ARC program for estimation of indications below the applied inspection distance have shown the initial assumptions applied in the original C* analysis that all tubes potentially contribute to leakage during to postulated SLB event is exceptionally conservative. These analyses support judgment of the cold leg C* distance for Palisades reaming at the value of 12.5 inches below the cold leg TTS, as defined by Reference 4.

Recommended +Point™[1] Probe Sampling for Cold Leg:

The most conservative estimate of cold leg PWSCC initiation concludes that 0.1% of the tubes would be affected by 48 EFPY in the more limiting of the two SGs (SGA). One indication would not be expected until the 1R32 outage (approximately 31 EFPY), two indications would not be expected until the 1R35 outage. For both SGs combined, 0.1% of the tubes are predicted to be affected at 53 EFPY. This projection was performed using a Weibull approximation with a bounding hot leg to cold leg improvement factor of 3.50.

A similar projection was performed for all hot leg tubesheet region outside diameter stress corrosion cracking (ODSCC) mechanisms. For both SGs combined, the time to initiate 2 cold leg ODSCC flaws (one in each SG) is 23.5 EFPY, or approximately the 1R27 outage. The time to achieve 0.1% affected (8 in each SG) is 40 EFPY, or beyond the operating license.

Prior W* applications have permitted the sampling of the cold legs once the failure analysis predicts indications may be initiated at the 0.1% affected level. This is at the earliest, 39 EFPY, or 1R38 for the limiting SG.

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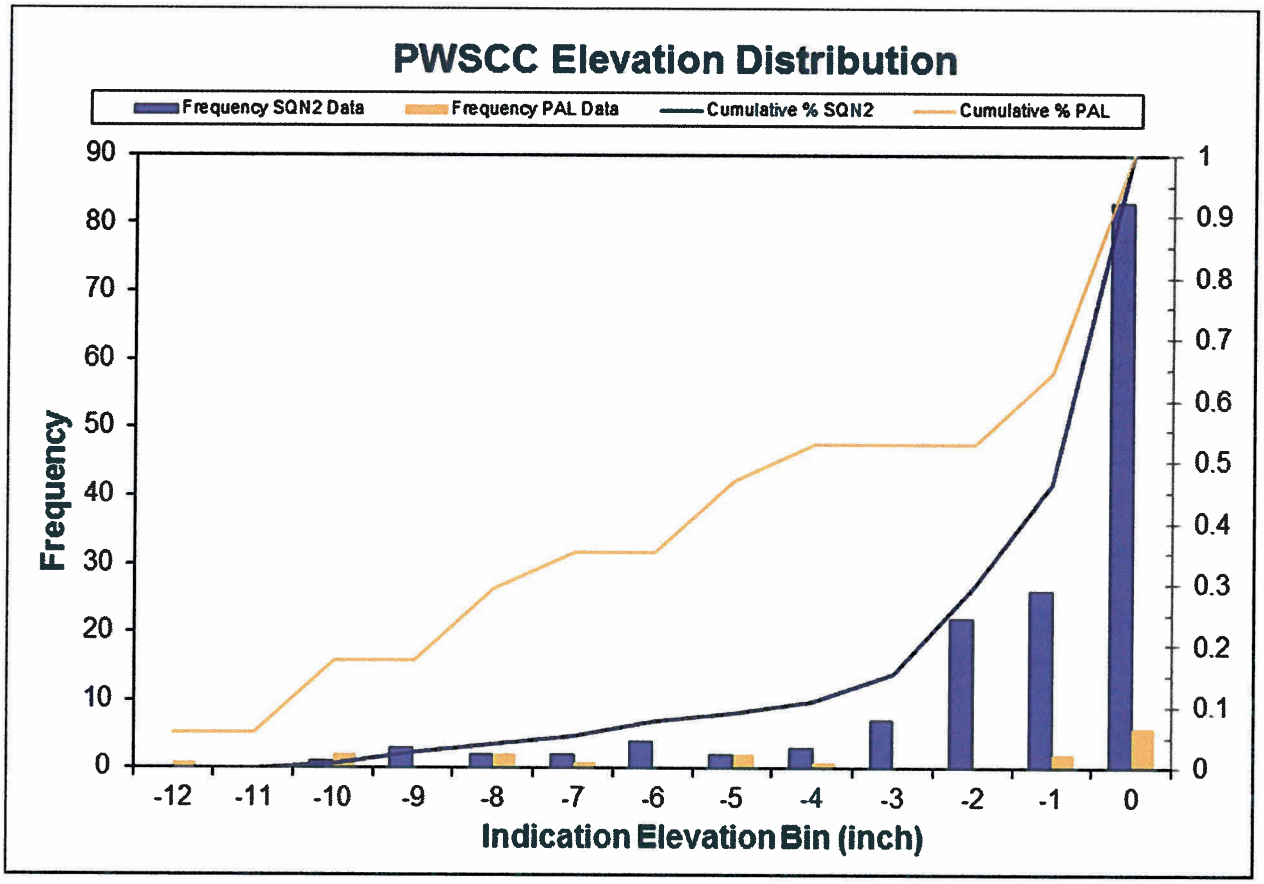
[1] +Point is a trademark of Zetec, Inc.

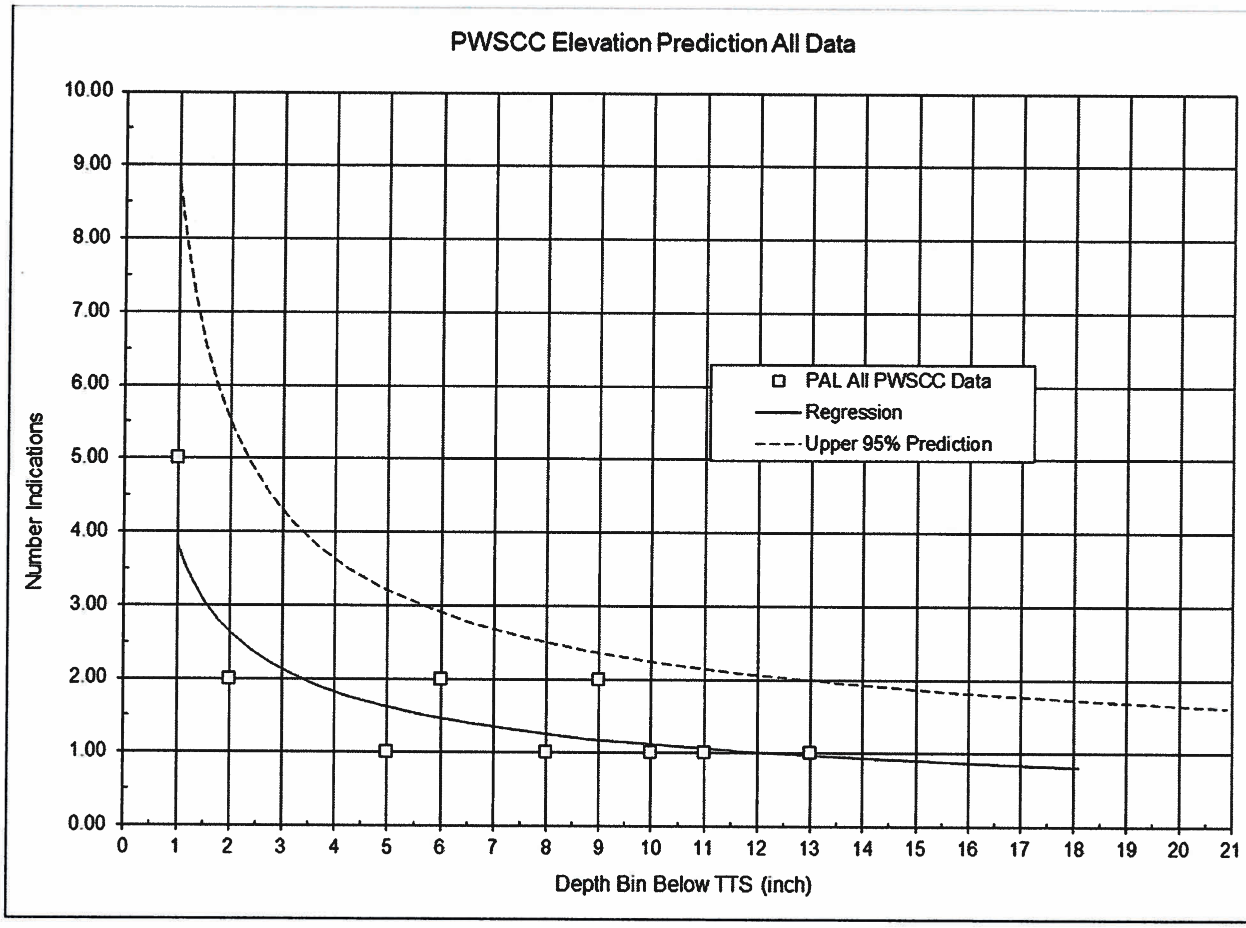
The first reporting of ODSCC at the hot leg top-of-tubesheet was the 2001 outage, with 7.57 EFPY accumulated since SG replacement. First expected occurrence on the cold leg can be approximated by multiplying the first reporting (7.57 EFPY) times the adjustment factor of 3.5, for an expected initiation by of 26.5 EFPY, which is slightly longer than the Weibull projection for both SGs.

Figure 1: PWSCC Elevation Distribution in Tubesheet Region for Plant SQN-2 and Palisades



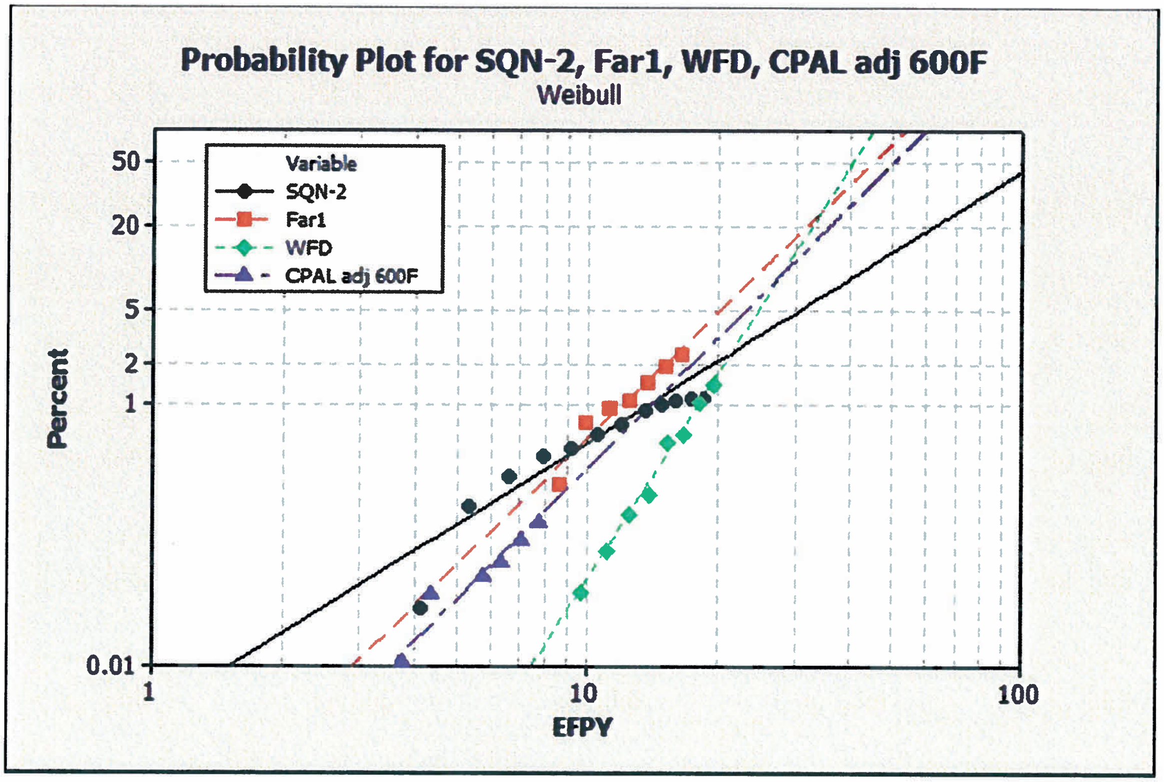
From file: cold leg Wstar initiation for PAL.xlsx.

Figure 2: Palisades PWSCC Elevation Prediction for Tubesheet Region



From File: PAL PWSCC Prediction.xlsx

Figure 3: Weibull Probability of Failure Estimate for Several Plants including Palisades (CPAL) Adjusted to a Hot Leg Temperature of 600°F



From file: MINITAB EXPLOSIVE EXPANSION PWSCC.mpj.

Figure 4: Comparison of C*, W*, and H* Room Temperature Leak Rate Testing

a,b,c

[ ]

Figure 5: Comparison of C*, W*, and H* Elevated Temperature Leak Rate Testing

a,b,c

[ ]

Figure 6: Comparison of C* and W* Elevated Temperature Leak Rate Testing

a,b,c

Figure 7: C* Leak Rate versus Joint Length at Lower 95% Confidence using Updated Uncertainty Analysis

a,b,c

REFERENCES:

1. LTR-CDME-06-80-P, Revision 1, "Palisades Tubesheet Inspection Depth," May 2006.

2. WEST-13-402, Rev. 1, "An Evaluation of the Statistical Variability in Coefficient of Thermal Expansion Properties of SA-508 and Alloy-600," Structural Integrity Associates, Inc., December 2008.

3. Nuclear Engineering and Design 143 (1993), "Residual Stresses Associated with the Hydraulic Expansion of Steam Generator Tubing into Tubesheets," March 1993.

4. SG-SGMP-10-4-P, Revision 1, "Palisades Cold Leg Tubesheet Inspection Depth, C*," February 2010.

5. LTR-SGMP-09-135, "Westinghouse Input to RAI Responses Supporting Sequoyah-2 Cold Leg W* License Amendment Request," September 2009.

6. ASME Boiler and Pressure Vessel Code. An American National Standard, Section II-D, "Materials," 2001 through 2003 Addenda.

7. Palisades Nuclear Plant Examination Technique Specification Sheet (ETSS) #2, "3-Coil RPC."

ATTACHMENT 9

Westinghouse Electric Company

Affidavit
CAW-16-4380

Application for Withholding Proprietary Information from Public Disclosure

February 23, 2016

8 pages follow

Westinghouse

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Proj Letter: CPAL-16-1

CAW-16-4380

February 23, 2016

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-SGMP-15-88, Rev. 1 P-Attachment, "Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis" (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC (Westinghouse), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-16-4380 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

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

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-16-4380, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

W. J. Snoody / FOR

James A. Gresham, Manager

Regulatory Compliance

CAW-16-4380
February 24, 2016

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Henry A. Sepp, am 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 my knowledge, information, and belief.

Henry A. Sepp, Director
CRE-Systems and Components Engineering

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-SGMP-15-88, Rev. 1 P-Attachment, "Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis" (Proprietary), for submittal to the Commission, being transmitted by Entergy Nuclear Palisades letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the technical justification for the C* Alternate Repair Criteria for explosively expanded tubes in the tubesheet region and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to support licensing the Alternate Steam Generator Tube Repair Criteria, C*.

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

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of supporting a response to possible NRC requests for additional information that would be necessary to license the C* Alternate Repair Criteria.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

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

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

Entergy Nuclear Palisades, LLC

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. One copy of "LTR-SGMP-15-88, Rev. 1 P-Attachment, "Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis" (Proprietary)

2. One copy of "LTR-SGMP-15-88, Rev. 1 NP-Attachment, "Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis" (Non-Proprietary)

Also enclosed is the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-16-4380, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-16-4380 and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.