Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530
Tel 269 764 2000

Anthony J. Vitale
Site Vice President

PNP 2016-037

June 7, 2016

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

SUBJECT: Response to - Palisades Nuclear Plant – Request for Additional Information Regarding the License Amendment Request for Implementation of an Alternate Repair Criterion on the Steam Generator Tubes (CAC No. MF7435)

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

REFERENCES:

1. Entergy Nuclear Operations, Inc. letter to NRC, PNP 2016-001, *License Amendment Request – Revision to the Requirements for Steam Generator Tube Inspections and Repair Criteria in the Cold Leg Tube Sheet Region*, dated March 3, 2016 (ADAMS Accession Number ML16075A103)

2. NRC letter to Entergy Nuclear Operations, Inc., *Palisades Nuclear Plant – Request for Additional Information Regarding the License Amendment Request for Implementation of an Alternate Repair Criterion on the Steam Generator Tubes (CAC No. MF7435),* dated May 11, 2016 (ADAMS Accession Number ML16130A076)

Dear Sir or Madam:

Entergy Nuclear Operations, Inc. (ENO) submitted a license amendment request (LAR) to the Nuclear Regulatory Commission (NRC) for approval to revise the requirements for steam generator (SG) tube inspections and repair criteria in the cold-leg tube sheet region (Reference 1). ENO received a letter from the NRC requesting additional information (RAI) associated with the LAR (Reference 2). ENO and the NRC held a conference call on May 5, 2016 to clarify the RAIs.

The enclosure to this letter provides ENO's response to the RAIs associated with the LAR. Included in the enclosure is an update to the no significant hazards (NSH) justification that was provided in Reference 1, Attachment 1, titled *Description and Evaluation of Requested Change*. The enclosure also contains updates to Reference 1, Attachment 2, titled *Proposed Changes to Palisades Plant Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications Pages*, and Reference 1, Attachment 3, titled *Page Change Instructions and Revised Pages for Palisades Plant Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications*. The updates to the NSH justification, Attachment 2, and Attachment

3 provided in the enclosure to this letter supersede what was previously submitted in Reference 1.

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 June 9, 2016.

Sincerely,

AJV/jpm

Enclosure: Response to Palisades Nuclear Plant – Request for Additional Information Regarding the License Amendment Request for Implementation of an Alternate Repair Criterion on the Steam Generator Tubes

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

ENCLOSURE

PNP 2016-037

Response to Palisades Nuclear Plant – Request for Additional Information
Regarding the License Amendment Request for Implementation of
an Alternate Repair Criterion on the Steam Generator Tubes

Attachments:

1. Updated No Significant Hazards Consideration

2. Updated Proposed Changes to Palisades Plant Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications Pages

3. Updated Page Change Instructions and Revised Pages for Palisades Plant Renewed Facility Operating License DPR-20 and Appendix A Technical Specifications

44 pages follow

Response to Palisades Nuclear Plant – Request for Additional Information Regarding the License Amendment Request for Implementation of an Alternate Repair Criterion on the Steam Generator Tubes

---

A request from the Nuclear Regulatory Commission (NRC) for additional information (RAI) regarding the license amendment request (LAR) to revise portions of the Palisades Nuclear Plant (PNP) Technical Specifications (TS), for implementation of an alternate repair criteria (ARC) on the steam generator (SG) tubes, was received by Entergy Nuclear Operations, Inc. via NRC letter dated May 11, 2016.

Nuclear Regulatory Commission (NRC) Request

By letter dated March 3, 2016 (Agencywide Documents Access and Management System Accession No. ML16075A103), Entergy Nuclear Operations, Inc. (the licensee) submitted a license amendment for Palisades Nuclear Plant (PNP). The proposed amendment would revise the PNP technical specifications, as they apply to the Steam Generator (SG) Program. Specifically, the licensee requested to implement an alternate repair criteria that invokes a C-Star inspection length (C), on a permanent basis for the cold-leg side of the SG's tube sheet.*

Based on its review of the amendment request, the U.S. Nuclear Regulatory Commission (NRC) staff has determined that additional information is required to complete the review. The requests for additional information below only pertain to the cold-leg; however, the licensee may want to consider revising the hot-leg requirements to prevent the TS from becoming overly complex.

<u>NRC Request SG ARC C* (RAI 1)</u>

In Section 5.1 of Attachment 1 (page 9), the tube-to-tubesheet (TTS) weld is described as a seal weld. As noted in Regulatory Issue Summary 2016-02, "Design Basis Issues Related to Tube-to-Tubesheet Joints in Pressurized-Water Reactor Steam Generators," dated March 23, 2016 (ADAMS Accession No. ML15169A543), the term "seal weld" does not always fully describe whether the weld was also qualified as a "structural weld."

> *Please clarify whether the TTS welds in the PNP SGs are qualified as structural welds and whether this qualification was in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.*

ENO Response (RAI 1 Part A)

The Palisades tube-to-tubesheet weld was analyzed as a structural load bearing member in the original steam generator design stress report. Stresses considered in the analysis are those resulting from primary side and secondary side pressures, operating transients, and displacements imposed on the weld due to tubesheet deflections. All stresses were found to be within the allowable values set forth in the ASME Code, Section III, NB-3000. In addition, a fatigue analysis according to the procedures outlined in NB-3222.4(d) of the applicable ASME code of construction at that time was performed; the maximum usage factor was determined to be less than 1.0 for the limiting location.

NRC Request SG ARC C* (RAI 1)

In Section 5.1 of Attachment 1 (page 9), the tube-to-tubesheet (TTS) weld is described as a seal weld. As noted in Regulatory Issue Summary 2016-02, "Design Basis Issues Related to Tube-to-Tubesheet Joints in Pressurized-Water Reactor Steam Generators," dated March 23, 2016 (ADAMS Accession No. ML15169A543), the term "seal weld" does not always fully describe whether the weld was also qualified as a "structural weld."

> *...If not, please discuss the design basis of the TTS joint to ensure structural integrity. Please discuss whether any of the qualification data for the TTS joint (if the joint is not a structural weld) is applicable to the C* methodology. If applicable, discuss its impact on your application for C*.*

ENO Response (RAI 1 Part B)

PNP tube-to-tube sheet (TTS) welds were analyzed as a structural load bearing member and therefore further discussion concerning the design basis of TTS joint structural integrity, qualification data applicability to the C* methodology, and impact on the application of C* is not necessary.

NRC Request SG ARC C* (RAI 2)

The proposed wording of TS Section 5.5.8c.2 addresses plugging tubes with flaws found within 12.5 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower. Since a flaw is generally regarded as a mark, fault or other imperfection, this requirement would require tubes with non-service-induced flaws (e.g., manufacturing burnish marks or over expansions) to be plugged. Although the industry may have different definitions for flaws in specific applications, those definitions are not part of the TS. Please clarify if plugging all non-service-induced flaws was your intent. If not, consider revising the TS for clarity.

ENO Response (RAI 2)

ENO's intended interpretation of the proposed TS section 5.5.8c.2 alternative repair criteria for the cold-leg tubesheet is that it is applicable to service-induced flaws only. This interpretation also applies to the TS section 5.5.8c.1 alternative repair criteria for the hot-leg tubesheet. To clarify this interpretation, ENO proposes to revise PNP TS Section 5.5.8c to more accurately define the type of flaws found within the C* length that would require plugging per the alternate repair criteria for both the hot-leg and cold-leg tubes. The proposed wording is as follows:

1. Tubes found by inservice inspection to contain service-induced 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.

2. Tubes found by inservice inspection to contain service-induced flaws within 13.67 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.

NRC Request SG ARC C* (RAI 3)

When the NRC staff reviewed the H alternate repair criteria, one of the concerns was that cracks could exist in the TTS welds. If H* was not applied to all tubes (i.e., if application of H* was optional, rather than mandatory), it was not clear to the NRC staff how the integrity of the TTS welds would be assured, since there was not a qualified inspection technique for the TTS welds. As a result, licensees that adopted H* applied it to all tubes in all of the steam generators at a given unit, rather than allowing an option that it may be applied.*

a. Please discuss whether C will be applied to all tubes on the cold-leg, rather than providing an option for it to be applied at PNP. The staff notes that if C* was approved but not implemented, any inspection program would have to consider the entire length of the tube within the tubesheet, not just the upper portion of the tube within the C* distance.*

ENO Response (RAI 3a)

ENO's interpretation of TS section 5.5.8c, *Provisions for SG tube repair criteria*, is that the alternative repair criteria apply to all tubes within the tubesheets in all SGs once the alternative repair criteria is approved. In order to clarify this interpretation, ENO proposes to revise PNP TS Section 5.5.8c by changing "may" to "shall." The proposed wording is as follows:

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 shall be applied as an alternate to the 40% depth based criteria:

   1. Tubes found by inservice inspection to contain service-induced 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.

   2. Tubes found by inservice inspection to contain service-induced flaws within 13.67 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.

NRC Request SG ARC C* (RAI 3)

When the NRC staff reviewed the H alternate repair criteria, one of the concerns was that cracks could exist in the TTS welds. If H* was not applied to all tubes (i.e., if application of H* was optional, rather than mandatory), it was not clear to the NRC staff how the integrity of the TTS welds would be assured, since there was not a qualified inspection technique for the TTS welds. As a result, licensees that adopted H* applied it to all tubes in all of the steam generators at a given unit, rather than allowing an option that it may be applied.*

b. *In the proposed TS 5.5.8d, periodic inspections are required to be performed from the TTS weld at the tube inlet (i.e., hot-leg), to the TTS weld at the tube outlet (i.e., cold-leg). During these inspections, if a crack were found between the C* depth and a TTS weld, TS 5.5.8d.3 requires an inspection for cracking in the next refueling outage, even though the tube with the crack would not require plugging in the current outage. Please discuss if TS 5.5.8d.3 is consistent with your proposed application of the C* alternate repair criteria. (The staff notes that licensees recently implementing similar alternate repair criteria to C*, have revised this section to limit inspection depth to that distance defined by the alternate repair criteria.)*

ENO Response (RAI 3b)

ENO's interpretation of TS section 5.5.8d is that the periodic inspection scope, which includes the percentage of the total population of SG tubes that are inspected each outage and the inspection length along each tube, is determined by the assessment of degradation. The degradation assessment would then credit the availability of the alternative repair criteria lengths in order to limit inspections in the SG tubesheets to the C* depth. In order to clarify this interpretation, ENO proposes to revise TS section 5.5.8d. The proposed wording is as follows:

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, 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, d.3, and d.4 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.

ENO's interpretation of TS section 5.5.8d.3 is that, since the alternate repair criteria only applies to tubes found by inspection to contain cracks within the C* depth then inspection scope expansion would also only apply to cracks found between the hot-leg and cold-leg C* depths. In order to clarify this interpretation, ENO proposes to revise TS section 5.5.8d.3. The proposed wording is as follows:

3. If crack indications are found in any SG tube from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, 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.

The above proposed revisions will clarify TS sections 5.5.8d and 5.5.8d.3 to ensure consistency between the alternate repair criteria and inspection scope expansion by limiting periodic inspections in the SG tubesheets to the C* depth and precluding inspection scope expansion if cracks are detected between the C* depth and a TTS weld.

NRC Request SG ARC C* (RAI 3)

When the NRC staff reviewed the H alternate repair criteria, one of the concerns was that cracks could exist in the TTS welds. If H* was not applied to all tubes (i.e., if application of H* was optional, rather than mandatory), it was not clear to the NRC staff how the integrity of the TTS welds would be assured, since there was not a qualified inspection technique for the TTS welds. As a result, licensees that adopted H* applied it to all tubes in all of the steam generators at a given unit, rather than allowing an option that it may be applied.*

c. *In the proposed TS 5.5.8d.5, there is a 100 percent sampling of cold-leg tubes when and if C* is implemented. The NRC staff notes that licensees adopting alternate repair criteria similar to C* (e.g., H* and F*) typically used an inspection strategy that was less than a 100 percent sample on the cold-leg, based on their degradation assessment and their performance-based technical specification requirement to maintain tube integrity. Given past precedent, discuss whether you would still like to retain the 100 percent sampling requirement.*

ENO Response (RAI 3c)

ENO proposed TS section 5.5.8d.5, cold-leg inspection scope expansion when the alternative repair criteria is implemented, for consistency with TS section 5.5.8d.4 currently in PNP's TS associated with the alternate repair criteria for the SGs hot-leg tubesheet. ENO interpreted TS section 5.5.8d.5 as requiring expansion to 100% tube inspection only when a crack was found within the C* depth. If cracking was not found in the tubesheets, then the degradation assessment would determine the SG tube inspection sample population. Since the proposed TS section 5.5.8d.5 could be interpreted to require 100% cold-leg sample size upon incorporation into PNP's TS and since the PNP SG hot-leg tubesheet tube sample population is currently at

100% inspection scope over the C* depth, ENO proposes to remove TS section 5.5.8.d.5 associated with the cold-leg alternate repair criteria from the license amendment request and retain TS section 5.5.8.d.4 associated with the hot-leg alternate repair criteria.

<u>NRC Request SG ARC C* (RAI 3)</u>

When the NRC staff reviewed the H alternate repair criteria, one of the concerns was that cracks could exist in the TTS welds. If H* was not applied to all tubes (i.e., if application of H* was optional, rather than mandatory), it was not clear to the NRC staff how the integrity of the TTS welds would be assured, since there was not a qualified inspection technique for the TTS welds. As a result, licensees that adopted H* applied it to all tubes in all of the steam generators at a given unit, rather than allowing an option that it may be applied.*

d. TS 5.5.8d states, "...requirements of d.1, d.2, and d.3, d.4, and d.5 below, ..." The staff notes that TS 5.5.8d contains an extra "and" between d.2 and d.3. Please discuss your plans to remove the redundant "and."

ENO Response (RAI 3d)

ENO proposes to remove the redundant "and" as requested. The proposed wording is as follows:

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, 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, d.3, and d.4 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.

<u>NRC Request SG ARC C* (RAI 4)</u>

TS 5.6.8i references monitoring tubes for displacement. The word displacement could be misinterpreted to mean rotation or bending. The slippage of concern associated with implementation of C is axial tube displacement. Was your intent to monitor and report all these displacements? If not, please clarify what you plan to monitor and report in the Steam Generator Tube Inspection Report.*

ENO Response (RAI 4)

ENO plans to monitor and report axial tube displacement only. To clarify this intent, ENO proposes to revise TS section 5.6.8i as follows:

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

NRC Request SG ARC C* (RAI 5)

The following questions pertain to Attachment 8 of the application, "Discussion of Applicability of H Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis," LTR-SGMP-15-88, Rev. 1 NP-Attachment, dated February 23, 2016.*

a. *Table 1 (page 4) provides cold-leg C* distances, ranging from 12.79 inches to 13.67 inches, for various coefficient of thermal expansion (CTE) conditions. As part of the basis for not increasing the previously calculated C* inspection distance of 12.5 inches, you referenced the current practice at PNP of inspecting one inch greater than the 12.5 inch inspection distance. It is possible that the tubesheet has a plus one sigma CTE and the tube has a minus one sigma CTE. Given that surveillance requirements are intended to assure the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met, please explain why the C* inspection distance on the cold-leg is proposed at 12.5 inches, or provide a revised value supported by the analysis of Attachment 8 of the application (e.g. 13.67 inches). If the likelihood of having specific circumstances occur (e.g., a tube with a specific coefficient of thermal expansion and a tubesheet with a specific coefficient of thermal expansion, at a specific location in the tube bundle) is used as the basis for your response to this question, the staff would expect that all significant parameters would be evaluated quantitatively and probabilistically (e.g., tube pullout data) and an appropriately conservative estimate of C* distance would be determined (e.g., .95 probability with a 95 percent confidence interval).*

ENO Response (RAI 5a)

ENO will revise the proposed C* inspection distance from 12.5 inches, which credited a statistical Monte Carlo analysis that determined the frequencies at which a simultaneous combination of coefficient of thermal expansion variances between the SG tube material and tubesheet material in the conservative direction could occur was insignificant, to 13.67 inches. This aligns with the results in Table 1 (page 4) of Westinghouse letter LTR-SGMP-15-88 NP-Attachment, Revision 1, *Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis*, by not limiting the C* inspection distance based on the Monte Carlo probabilistic analysis. The revised C* distance is reflected in the proposed TS section 5.5.8c.2 as follows:

2. Tubes found by inservice inspection to contain service-induced flaws within 13.67 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.

NRC Request SG ARC C* (RAI 5)

The following questions pertain to Attachment 8 of the application, "Discussion of Applicability of H Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis," LTR-SGMP-15-88, Rev. 1 NP-Attachment, dated February 23, 2016.*

b. On page 5, it is stated that the only potential leakage source is from postulated indications below the inspection distance, since indications detected within the C distance must be plugged. Although this may be true most of the time, unexpected conditions could arise (e.g., missed indications or indications below the threshold of detectability within the C* region). Please discuss your plans to assess the indications detected within the C* distance to confirm that they do not pose an accident induced leakage concern.*

ENO Response (RAI 5b)

Indications detected within the C* distance will be evaluated in the steam generator condition monitoring report and operational assessment report for leakage integrity and structural integrity using standard practices outlined in the Electric Power Research Institute (EPRI) Steam Generator Tube Integrity Guidelines. If a leakage potential is realized the estimated leakage contribution will be considered against the accident induced leakage performance criterion (AILPC) value for Palisades.

ENCLOSURE
ATTACHMENT 1

Updated No Significant Hazards Consideration

5 pages follow

The Entergy Nuclear Operations, Inc. (ENO) response to requests for additional information (RAI) 2, 3, 4, and 5 of this enclosure includes changes to the license amendment request (LAR) proposed technical specifications (TS) wording. Some of the revised wording is applicable to the hot-leg tubesheet as well as the cold-leg tubesheet and therefore impacts the no significant hazards consideration that was provided in the LAR in Reference 1 (PNP 2016-001). A summary of the proposed TS wording changes is provided below.

In response to RAI 2, ENO has further clarified the interpretation of the word “flaws” by adding the descriptor “service-induced” prior to the word “flaws” in the first sentence of TS sections 5.5.8c.1 and 5.5.8c.2.

In response to RAI 3, ENO proposed the following changes:

TS Section	Proposed TS Wording Change	Basis
5.5.8c	Change the word “may” to “shall.”	To more accurately describe application of the alternate repair criteria.
5.5.8d	Change “tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet” to “12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower.”	To more accurately describe the length along the SG tubes that inspection occurs once the alternate repair criteria is in place for both the hot-leg and cold-leg tubesheets.
5.5.8d.3	Add “from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of cold-leg tubesheet, whichever is lower” after “If crack indications are found in any SG tube.”	To clarify SG inspection expansion requirements and improve consistency with TS section 5.5.8d.
5.5.8d.5	Delete “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.”	To remove redundancy with TS section 5.5.8d.3 inspection expansion requirements and preclude the interpretation that 100% inspection scope is required in the cold-leg tubesheet once the alternate repair criteria is approved.
5.5.8d	Remove the redundant “and” in fifth sentence between d.2 and d.3.	Editorial change

In response to RAI 4, ENO has further described “tube displacement (slippage)” by adding the word “axial” before “displacement” in TS section 5.6.8i.

In response to RAI 5, ENO has revised the cold-leg C* inspection distance from 12.5 inches to 13.67 inches. Increasing the C* inspection distance to 13.67 inches removes reliance on a probabilistic analysis that was credited as a basis for limiting the C* inspection distance to 12.5 inches. This change is reflected in TS section 5.5.8c.2.

Therefore, based on the RAI responses, an updated no significant hazards consideration which reflects the updated TS wording is provided below. It supersedes in its entirety the no significant hazards consideration provided in the LAR.

Updated No Significant Hazards Consideration

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 5.5.8c.2, alternate repair criteria (ARC), based on a C-star (C*) inspection length, on a permanent basis for the cold-leg side of the Palisades Nuclear Plant (PNP) steam generators' (SG) tube sheet. To clarify intent and improve interpretation of PNP TS, which has previously incorporated ARC for the hot-leg side of the SG's tube sheet under Amendment 225, the proposed wording for TS sections 5.5.8c.1, 5.5.8d, 5.5.8d.3, and 5.6.8i has been changed.

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 section 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 is consistent with 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 section 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.

In 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*, the 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 13.67 inches. 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* length for the SG cold-legs provides 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 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 13.67 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 the 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

ENCLOSURE
ATTACHMENT 2

Updated

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)

NOTE:

The pages in this attachment replace all pages of Attachment 2 to Entergy Nuclear Operations, Inc. letter to NRC, PNP 2016-001, *License Amendment Request – Revision to the Requirements for Steam Generator Tube Inspections and Repair Criteria in the Cold Leg Tube Sheet Region*, dated March 3, 2016 (ADAMS Accession Number ML16075A103)

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~~shall be applied as an alternate to the 40% depth based criteria:

1. Tubes found by inservice inspection to contain service-induced 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 service-induced flaws within 13.67 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~~ 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, 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 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~~,~~ from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of cold-leg tubesheet, whichever is lower, 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

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

(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.

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.

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 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.

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 $\geq$ 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 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 Programs and Manuals

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 axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

ENCLOSURE
ATTACHMENT 3

Updated

Page Change Instructions and Revised Pages for

Palisades Plant
Renewed Facility Operating License DPR-20

and

Appendix A
Technical Specifications

NOTE:

The pages in this attachment replace all pages of Attachment 3 to Entergy Nuclear Operations, Inc. letter to NRC, PNP 2016-001*, License Amendment Request – Revision to the Requirements for Steam Generator Tube Inspections and Repair Criteria in the Cold Leg Tube Sheet Region*, dated March 3, 2016 (ADAMS Accession Number ML16075A103)

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

REMOVE	INSERT
Page 3	Page 3

Appendix A, Technical Specifications

REMOVE	INSERT
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

(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. 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, 5.5.8 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 shall be applied as an alternate to the 40% depth based criteria:

1. Tubes found by inservice inspection to contain service-induced 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 service-induced flaws within 13.67 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 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, 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, d.3, and d.4 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 from 12.5 inches below the bottom of the hot-leg expansion transition or top of the hot-leg tubesheet, whichever is lower, to 13.67 inches below the bottom of the cold-leg expansion transition or top of the cold-leg tubesheet, whichever is lower, 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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

d. Provisions for SG tube inspections. (continued)

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.

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.

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%

Page renumbered

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.

Page renumbered

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.

Page renumbered

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

Page renumbered

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.

Page renumbered

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 <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.

Page renumbered

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.

Page renumbered

5.5 Programs and Manuals

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.

Page renumbered

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 axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.