UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 19, 2016

Vice President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT RE: REVISION TO THE REQUIREMENTS FOR STEAM GENERATOR TUBE INSPECTIONS AND REPAIR CRITERIA IN THE COLD LEG TUBE SHEET REGION (CAC NO. MF7435)

Dear Sir or Madam:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 261 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant (PNP). The amendment approves changes to the PNP technical specifications (TS) in response to your application dated March 3, 2016, as supplemented by letter dated June 7, 2016.

Specifically, the amendment approves the licensee's request to implement an alternate repair criteria (ARC) called C-star, for the portion of the steam generator (SG) tubes within the cold-leg tubesheet. In addition, the amendment approves the license's request to clarify the intent and improve the wording of the TS regarding the previously incorporated ARC for the hot-leg side of the SG's tubesheet. This was previously approved by letter dated May 31, 2007, and Amendment No. 225.

A copy of our related safety evaluation is provided in Enclosure 2. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Jennivine K. Rankin, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 261 to DPR-20
2. Safety Evaluation

cc w/encls: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-255

PALISADES NUCLEAR PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 261
License No. DPR-20

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee), dated March 3, 2016, as supplemented June 7, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to the license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-20 is hereby amended to read as follows:

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 261, 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. This license amendment is effective as of the date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License No. DPR-20 and the Technical Specifications

Date of Issuance: December 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 261

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

Replace the following page of the Renewed Facility Operating License No. DPR-20 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

REMOVE	INSERT
Page 3	Page 3

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
Pages 5.0-12 to 5.0-22	Pages 5.0-12 to 5.0-22
Page 5.0-28	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. 261, 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, August

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%

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_2O)	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 $\geq$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 <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.

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.

5.6 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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 261 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

ENTERGY NUCLEAR OPERATIONS, INC.

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By application dated March 3, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16075A103), as supplemented by letter dated June 7, 2016 (ADAMS Accession No. ML16159A230), Entergy Nuclear Operations, Inc. (ENO, the licensee), submitted a licensee amendment request (LAR) to implement an alternate repair criteria (ARC) called C-star (C*) for the portion of the steam generator (SG) tubes within the cold-leg tubesheet. In addition, the LAR further clarifies wording in the Palisades Nuclear Plant (PNP) technical specifications (TS) that apply to the C* SG ARC that was already approved by the U.S. Nuclear Regulatory Commission (NRC or Commission) for use on the SG hot-leg tubesheet. The proposed changes revise the PNP TS Sections 5.5.8, "Steam Generator (SG) Program," and 5.6.8, "Steam Generator Tube Inspection Report." The implementation of C* will result in the licensee not having to inspect the lower portion of the SG tubes within the cold-leg tubesheet, since leakage from flaws in this region would be acceptable.

The NRC staff published its original no significant hazards consideration determination in the *Federal Register* on June 7, 2016 (81 FR 36604). A revised no significant hazards consideration determination was published in the *Federal Register* on August 2, 2016 (81 FR 50747), to consider a revised description of the amendment request and associated changes to the no significant hazards consideration determination provided in the supplemental letter dated June 7, 2016.

2.0 REGULATORY EVALUATION

The tubes in a SG function as an integral part of the reactor 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, the NRC requires licensees to perform periodic inservice inspections of SG tubes. These inspections detect, in part, flaws in the tubes resulting from interaction with the SG operating environment, including both primary and secondary coolant. Inservice inspections may also provide a means of

Enclosure 2

characterizing the nature and cause of any tube flaws so that corrective measures can be taken. Tubes with flaws that exceed the tube plugging limits specified in a plant's TS are removed from service by plugging. The plant TS provide the acceptance criteria related to the results of SG tube inspections.

The requirements for the inspection of SG tubes are intended to ensure that this portion of the reactor coolant system maintains its integrity. Tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis, including the TS. Tube integrity includes both structural and leakage integrity. 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. These limits ensure that radiological dose consequences associated with any leakage are within acceptable limits and they limit the frequency of SG tube ruptures.

The following explains the applicability of General Design Criteria (GDC) for PNP. The construction permit for PNP was issued by the Atomic Energy Commission (AEC) on March 14, 1967, and an Interim Provisional Operating License was issued by the AEC on March 24, 1971. The plant GDCs are discussed in the Updated Final Safety Analysis Report (UFSAR) Chapter 5.1, "General Design Criteria," with more details given in the applicable UFSAR sections. The AEC published the final rule that added Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971, with the rule effective on May 21, 1971. As discussed in the NRC's Staff Requirements Memorandum for SECY-92-223, dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC which constitute the licensing bases for PNP are those in the UFSAR.

The licensee has made changes to the facility over the life of the plant that may have invoked the final GDC. The extent to which the final GDC have been invoked can be found in specific sections of the UFSAR and in other plant-specific design and licensing basis documentation.

In reviewing requests of this nature, the NRC staff verifies that a methodology exists that maintains the structural and leakage integrity of the tubes consistent with the plant design and licensing basis. This includes verifying that the applicable GDC, e.g., GDC 14, "Reactor coolant pressure boundary," and GDC 32, "Inspection of reactor coolant pressure boundary," contained in Appendix A to 10 CFR Part 50 and the performance criteria in the plants TS are satisfied. As stated in Chapter 5.1 of the UFSAR, the plant design meets the criteria specified in GDC 14 and GDC 32.

The NRC staff's evaluation also includes verifying that a methodology exists for determining the amount of primary-to-secondary leakage that may occur during design-basis accidents (DBAs). The amount of leakage is limited to ensure that offsite and control room dose criteria are met. The radiological dose criteria are specified, in part, in 10 CFR Part 100, "Reactor Site Criteria," in 10 CFR Part 50.67, "Accident source term," and in GDC 19, "Control room" of Appendix A to 10 CFR Part 50. No accident analyses for PNP are being changed because of the proposed amendment and, thus, no radiological consequences of any accident analysis are being changed. The proposed changes maintain the accident analyses and consequences that the

NRC has reviewed and approved for the postulated DBAs for SG tubes.

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TS are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

10 CFR 50.36(c)(5), "Administrative controls," includes "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee for operating the facility in a safe manner including the SG program, are listed in the administrative controls section of the TSs. The licensee's proposed changes concern the PNP TS Administrative Controls Section 5.5.8, "Steam Generator (SG) Program," and 5.6.8, "Steam Generator Tube Inspection Report."

3.0 TECHNICAL EVALUATION

3.1 Background

PNP is a two-loop, Combustion Engineering (CE) designed plant operating with a hot-leg temperature of 583 degree Fahrenheit (°F) and a cold-leg temperature of 537 °F. The two SGs currently installed are CE Model 2530 replacement SGs that were placed into operation in the fall of 1990. Each SG contains 8,219 mill-annealed Alloy 600 tubes with a nominal outside diameter of 0.75 inches and a nominal wall thickness of 0.042 inches. The tubes were explosively expanded at both ends for the full depth of the 20.5 inches thick tubesheet, using a process called "explansion." Near the top of the tubesheet, the transition from the expanded portion of the tube to the unexpanded portion of the tube is referred to as the expansion transition. The tube bundle is held in place by a stainless steel structure that is comprised of horizontal lattice-type supports, vertical straps and diagonal straps. The tube bundle is constructed as follows: the tubes in rows 1-18 have 180 degree U-bends, the tubes in rows 19-138 have two 90 degree bends with a horizontal run between the bends, and there are 195 columns of tubes.

A tube-to-tubesheet joint consists of a tube (which has been inserted into and then expanded against the wall of a hole drilled through the tubesheet), the tubesheet, and a tube-to-tubesheet weld (which is located at the end of the tube). Each tube has two tube-to-tubesheet joints, one at each end of the tube. Typically, the tube-to-tubesheet joints in a SG are designed as welded joints rather than friction joints. That is, the tube-to-tubesheet weld itself is designed as the pressure boundary element that transmits the entire differential pressure load from the tube to the tubesheet, with no credit taken for the friction developed between the expanded tube and tubesheet. In addition, the weld makes the joint leak tight.

The existing inspection requirements in the plant TS do not take into account the reinforcing effect of the tubesheet on the external surface of the expanded tube. Nonetheless, the presence of the tubesheet constrains the tube and complements tube integrity in that region by preventing tube deformation beyond the expanded outside diameter of the tube. The resistance to both tube rupture and tube collapse is significantly enhanced by the tubesheet reinforcement.

In addition, the proximity of the tubesheet to the expanded tube significantly reduces the leakage from any through-wall flaw.

Based on these considerations, power reactor licensees have proposed, and the NRC has approved, ARC for flaws located in SG tubes that are contained in the lower portion of the tube within the tubesheet, when these flaws are a specific distance below the bottom of the expansion transition (BET) or the top of the tubesheet (TTS) whichever is lower.

The C* methodology defines a distance, referred to as the C* distance, such that any type or combination of flaws below this distance (including flaws in the tube-to-tubesheet weld) are considered acceptable. That is, even if inspections below the C* distance identify flaws, the regulatory requirements pertaining to tube structural and leakage integrity would be met, provided there were no significant flaws within the C* distance. The C* distance is determined by calculating the amount of non-degraded tubing needed to ensure the tube will not pull out of the tubesheet and that the amount of leakage from flaws below the C* distance is limited (i.e., within acceptance limits). The C* distance is measured down from the TTS or the BET whichever is lower.

Nondestructive examination (NDE) uncertainties are accounted for in determining the C* distance. These uncertainties include, but are not limited to, the uncertainties in determining the location of the BET and the inspection distance of the tube below the BET (i.e., the C* distance).

The C* analysis presented in Westinghouse Commercial Atomic Power report, WCAP-16208-NP, Revision 1 (and the associated proprietary version), "NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansion," dated May 31, 2005 (ADAMS Accession No. ML051520417), used non-plant-specific primary and secondary system pressures and temperatures to determine the C* distance for tubing within the hot-leg side of the SG tubesheet for a number of plants with CE SGs. Because the operating conditions identified in WCAP-16208-NP, Revision 1, did not bound the PNP operating conditions, PNP used a revised C* distance for the SG hot-leg in the LAR they submitted on May 30, 2006 (ADAMS Accession No. ML061560406).

The C* analysis considers the forces acting to pull the tube out of the tubesheet (i.e., from the differential pressure between the primary and secondary sides of the tube) and the forces acting to keep the tube in place. These latter forces are a result of friction and the forces arising from: (1) the residual preload from the explansion process, (2) the differential thermal expansion between the tube and the tubesheet, and (3) internal pressure in the tube within the tubesheet. In addition, the effects of tubesheet bow, due to pressure and thermal differentials across the tubesheet, were considered since this bow causes dilation of the tubesheet holes from the secondary face to approximately half the thickness of the tubesheet and reduces the ability of the tube to resist pullout. The amount of tubesheet bow varies as a function of radial position with locations near the periphery and near the stay cylinder experiencing less bow. The effects of tubesheet hole dilation were analyzed using the worst-case hole (location) in the tubesheet.

Because temperature affects the tightness of the tube-to-tubesheet joint, the difference in temperature between the hot-leg and cold-leg portions of the tubesheet must be accounted for (e.g., through calculation of different C* distances for the hot-leg and cold-leg).

3.2 Evaluation of Proposed Changes

The current TS, the licensee's proposed changes, and the NRC staff evaluation of the proposed changes are provided below.

3.2.1. TS 5.5.8c

The proposed amendment revises TS 5.5.8c to indicate that the ARC shall be applied as an alternate to the 40 percent depth based criteria.

Current TS 5.5.8c states:

> 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:

Revised TS 5.5.8c would state:

> 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:

Evaluation of TS 5.5.8c

The existing TS 5.5.8c specifies that tubes containing flaws equal to or greater than 40 percent in depth shall be plugged, unless the flaws meet the requirements of the ARC in TS 5.5.8c.1. The ARC currently in TS 5.5.8c.1 may be applied to flawed tubes within the hot-leg tubesheet only. Tubes in the hot-leg, with flaws within 12.5 inches below the TTS or the BET, whichever is lower, shall be plugged; tubes with flaws below this elevation may remain in service. The proposed amendment revises TS 5.5.8c. to specify that the C* ARC shall be applied as an alternate to the 40 percent depth-based criteria for tube plugging. The NRC staff previously approved C* for use on the hot-leg; therefore, requiring it to be implemented is acceptable.

3.2.2. TS 5.5.8c.1

The proposed amendment revises TS 5.5.8c.1 to clarify that the ARC applies to tubes with "service-induced flaws," as opposed to the current wording of "flaws."

Current TS 5.5.8c.1 states:

> 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

Revised TS 5.5.8c.1 would state:

> 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.

Evaluation of TS 5.5.8c.1

This change clarifies the nature of the flaws to which the C* ARC will be applied and is in alignment with the licensee's SG Program; therefore, the NRC staff finds this change acceptable. The addition of a period at the end of 5.5.8c.1 is an administrative change that the NRC staff finds acceptable.

3.2.3. TS 5.5.8c.2

The proposed amendment adds TS 5.5.8c.2 in its entirety, which specifies that the C* ARC shall be applied to tubes in the cold-leg tubesheet, and uses a C* distance of 13.67 inches.

New TS 5.5.8c.2 would state:

> 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.

Evaluation of TS 5.5.8c.2

This change is in accordance with the technical basis submitted by the LAR and the C* methodology, which the NRC staff reviews in Sections 3.3 and 3.4 of this safety evaluation; therefore, the NRC staff finds its use acceptable.

3.2.4. TS 5.5.8d

Implementing the C* methodology also eliminates the need to inspect the portion of the tube within the hot-leg and cold-leg tubesheet regions below the C* distance, since the inspection provision in TS 5.5.8d requires that tubes be inspected with the objective of detecting flaws that may satisfy the applicable tube repair criteria. With no repair criteria to satisfy, the portions of the tube below the C* distance are not subject to the inspection provision and TS 5.5.8d is revised with C* distances of 12.5 and 13.67 inches for the hot-leg and cold-leg, respectively.

Current TS 5.5.8d states, in part:

> 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 below, the inspection scope...

Revised TS 5.5.8d would state, in part:

> 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...

Evaluation of TS 5.5.8d

The NRC staff finds these changes acceptable, as reviewed in Sections 3.3 and 3.4 of this safety evaluation, for the cold-leg tubesheet. The changes for the hot-leg were previously approved by the NRC by letter dated May 31, 2007. The staff notes that the different inspection distances arise primarily because of different temperatures on the hot-leg and cold-leg.

3.2.5. TS 5.5.8d.3

TS 5.5.8d.3 is revised to be consistent with 5.5.8d.

Current TS 5.5.8d.3 states:

> 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.

Revised TS 5.5.8d.3 would state:

> 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 (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.

Evaluation of TS 5.5.8d.3

The NRC staff finds these changes acceptable, as reviewed in Sections 3.3 and 3.4 of this safety evaluation, for the cold-leg tubesheet. The changes for the hot-leg were previously approved by the NRC by letter dated May 31, 2007.

3.2.6. TS 5.5.8d.4

TS 5.5.8d.4 is revised by deleting a period in reference to TS 5.5.8c.1 in two places, which accurately reflects the section numbering of the reference.

Current TS 5.5.8d.4 states:

> 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.

Revised TS 5.5.8d.4 would state:

> 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.

Evaluation of TS 5.5.8d.4

These are administrative changes the NRC staff finds acceptable. In addition, subsequent TS pages were renumbered due to the changes described in Sections 3.2.1 through 3.2.6 of this safety evaluation. The staff finds the renumbering of the TS pages to be administrative and acceptable.

3.2.7. TS 5.6.8i

The proposed amendment adds TS 5.6.8i in its entirety, which specifies that the licensee will monitor the SGs for tube axial displacement and report such findings if slippage is noted, along with the corrective actions taken.

New TS 5.6.8i would state:

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

Evaluation of TS 5.6.8i

Under the reporting requirements of TS 5.6.8 for PNP, the licensee is required to submit specific information to the NRC within 180 days after the reactor coolant system reenters Mode 4 following a SG tube inspection. Among other things, 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 that are within the hot-leg and cold-leg C* distances. The proposed TS 5.6.8i would report the results of monitoring for tube axial displacement (slippage). The NRC staff has reviewed the reporting requirements and finds that they are sufficient to allow the staff to monitor implementation of the proposed amendment and verify that operating experience continues to be conservative relative to the assumptions made in the amendment. As a result, the NRC staff finds the reporting requirements acceptable.

3.3 NRC Evaluation of Tube Structural Integrity

The proposed amendment will permit tubes with flaws to remain in service; therefore, the licensee must demonstrate that the tubes returned to service using the C* methodology will maintain adequate structural integrity for the period between inspections. Tube rupture and the pullout of a tube from the tubesheet are the two potential modes of structural failure considered for tubes returned to service under the C* methodology.

After implementing the C* criteria, a tube flaw would need to grow above the tubesheet's secondary face in order to rupture. If the entire flaw remains within the tubesheet, the reinforcement provided by the tubesheet will prevent tube rupture. The C* methodology proposed by PNP requires an inspection of tubes for the applicable hot-leg or cold-leg C* distance and the plugging of any tubes found with flaws within the applicable C* distance. Therefore, after inspection, any known flaws remaining in service will be located a minimum of 12.5 inches below the top of the hot-leg side of the tubesheet (previously approved by Amendment No. 225 in May 2007) and 13.67 inches below the top of the cold-leg side of the tubesheet. Industry operating experience shows flaw growth rates within the tubesheet are well below that necessary to propagate a flaw from below the C* distance to above the TTS in one operating interval. Thus, tube burst is precluded for these flaws due to the reinforcement provided by the surrounding tubesheet. There is a potential that there are flaws that are not detected within the C* distance. Operating experience indicates it is unlikely for these flaws to grow above the TTS and become susceptible to rupture in one operating interval.

The other postulated structural failure mode for tubes remaining in service using the C* methodology is pullout of the tube from the tubesheet, due to axial loading on the tube. The differential pressure between the primary side and the secondary side of the SG imparts an axial load into each tube that is counteracted by the tube-to-tubesheet joint. Axial tube loading during normal operating conditions can be significant; however, the peak postulated loading occurs during events involving a depressurization of the secondary side of the SG (e.g., main steam line break (MSLB)). The presence of flaws within a SG tube decreases the load bearing capability of the affected tube. If a tube becomes sufficiently degraded, these loads could lead to an axial separation of the tube.

Resistance to tube pullout is provided by the interference fit created during the tube explosive expansion process. In addition, the differential thermal expansion between the tube and the tubesheet, and the internal pressure of the tube, both tighten the interference fit between the tube and the tubesheet, to further resist tube pullout. Conversely, resistance to tube pullout is reduced by tubesheet bow, which causes the tubesheet holes to dilate near the top of the tubesheet.

In addition, tube pullout is restricted by the tube bundle upper support structure. The upper support structure design is such that vertical movement of tubes is limited by supports or neighboring tubes for all tubes except those on the periphery of the tube bundle. The proposed inspection distance does not take credit for restriction to tube movement inherent to the SG design.

The analysis supporting the licensee's proposed modifications to the tube inspection requirements addressed the limiting conditions necessary to maintain adequate structural integrity of the tube-to-tubesheet joint. Specifically, the tube must not experience excessive displacement relative to the tubesheet under bounding loading conditions with appropriate factors of safety considered. For C*, the most limiting condition for structural integrity is maintaining a margin of three against the axial loads experienced during normal operation. The original analysis documented in WCAP-16208-NP for the hot-leg side of the SG was previously evaluated by the NRC and approved via the letter, "Palisades Nuclear Plant - Issuance of Amendment Re: Tubesheet Inspection Depth for Steam Generator Tube Inspections (TAC No. MD2125)," dated May 31, 2007.

In the current LAR, the licensee supplemented the analysis documented in WCAP-16208-NP, Revision 1, for use on the cold-leg side of the SGs, with the following documents:

- SG-SGMP-10-4-NP, Revision 1, "Palisades Cold Leg Tubesheet Inspection Depth, C*," dated February 2010.
- LTR-SGMP-15-88, Revision 1 NP-Attachment, "Discussion of Applicability of H* Lessons Learned, If Applicable to the Palisades Nuclear Plant Cold Leg C* Analysis," dated February 23, 2016.

These analyses compensated for the lower temperature on the cold-leg side of the SGs (532 °F versus the 583 °F hot-leg temperature) and used updated material property values for Young's Modulus and the coefficient of thermal expansion for the tubes and tubesheet. The analyses showed that the previously calculated pullout distance of 5.25 inches would not be expected to be affected by more than approximately one inch, and was still bounded by the leakage-based C* distance of 13.67 inches being proposed for the cold-leg side of the SGs. In addition, the analysis provided used conservative assumptions in the tube pullout analysis such as: worst-case tube dilation, use of limiting pullout data, and use of a 95 percent upper bound on NDE uncertainty.

In summary, the NRC staff concludes that the proposed tube-to-tubesheet joint length (or inspection distance) is acceptable to ensure structural integrity of the tubesheet joint. This conclusion is based on numerous factors including: the presence of the tubesheet that precludes tube burst; past inspection results that indicate flaws are detected early enough to prevent structurally significant flaws from developing within the C* inspection distance; the conservative assumptions in the tube pullout analysis; the confirmation through testing that the analytical adjustments for pressure and temperature are supported; the restriction to tube pullout provided by the tube bundle upper support structure; and the large margin between the proposed C* distance and the length of tubing needed to restrict pullout.

3.4 NRC Evaluation of Tube Leakage Integrity

In assessing leakage integrity of a SG under postulated accident conditions, the leakage from all sources (i.e., all types of flaws at all locations and all non-leak tight repairs) must be assessed. The combined leakage from all sources is limited to below a plant-specific limit based on radiological dose consequences, with a maximum of 1 gallon per minute (gpm) per SG, unless the NRC staff has approved an exception for specific types of degradation at specific locations (which gives consideration to severe accident risk). The licensee reported this plant-specific limit for PNP is 0.3 gpm per SG. This limit is referred to as the "accident-induced leakage limit."

As part of the C* methodology, the licensee restricts the amount of primary-to-secondary leakage from the tube-to-tubesheet joints to 0.2 gpm per SG under the most limiting design basis accident conditions, which is an MSLB. That is, the inspection distances required by this proposal (12.5 inches in the hot-leg and 13.67 inches in the cold-leg) were determined based on ensuring the leakage from implementation of C* would contribute 0.2 gpm per SG to the total leakage under MSLB conditions. The 0.2 gpm contribution to the accident-induced leakage from tubes remaining in service according to the C* methodology is below the overall plant-specific leakage limit of 0.3 gpm per SG.

The licensee's method for determining the amount of leakage from flaws within the tubesheet region considered flaws located both within and below the C* distance. For flaws located within the C* distance, no leakage is anticipated since the proposed TSs state that all degradation in this region will be plugged on detection. As a result, the only flaws expected within the C* distance would be either newly initiated or undetected (e.g., below the threshold of detection). These flaws typically do not grow in one operating cycle to the extent that they would leak during post-accident conditions. Although no leakage is expected from flaws within the C* distance, the licensee indicated they will assess potential leakage from such flaws as part of their assessments.

Since the C* methodology does not require inspections below the C* distance, there is a potential that flaws which could leak will exist below this elevation in each tubesheet region. As a result, the licensee developed a methodology for determining the amount of accident-induced primary-to-secondary leakage from flaws in this region of the tubesheet. This methodology and the NRC staff's review of this methodology are discussed below.

The amount of leakage from flaws below the inspection distance depends on the number of flaws, the locations of the flaws, and the severity of the flaws. The methodology developed by the licensee assumes that every tube has a 360-degree circumferential, 100 percent through-wall flaw (i.e., a tube sever) at the bottom of the C* inspection distance in both the hot-leg and cold-leg tubesheet regions. In this analysis, the assumed number of inservice tubes per SG (7,846) is greater than the actual limiting number of inservice tubes (7,826), and is therefore conservative with respect to leakage. Given past plant-specific and industry operating experience, the staff considers the assumption that all tubes contain circumferential, through-wall flaws at the C* distance to be conservative.

The NRC staff also considers assuming only one flaw per tube acceptable since leakage from flaws in the lower half of the tubesheet would not be expected to contribute significantly to leakage given (1) the length of the tube-to-tubesheet crevice, (2) tubesheet bow in the lower region of the tubesheet tends to increase the resistance to leakage (since tubesheet bow in the lower region of the tubesheet tends to make most holes contract rather than dilate), and (3) the amount of leakage from the portion of the tube within the tubesheet region will predominantly be a function of the flaw nearest the top of the tubesheet (i.e., at 12.5 or 13.67 inches below the top of the tubesheet as assumed by the licensee).

Assuming flaws are present below the C* distance, the licensee's methodology determined the amount of leakage from the flaws left in service with the C* criteria using a combination of laboratory leak test data and analysis. Leak tests were performed on 0.75-inch outside diameter Alloy 600 mill annealed tube samples expanded into an 8-inch thick carbon steel collar to represent SG tubes explosively expanded into tubesheet holes. Two different finishes for the holes in the carbon steel collars were tested, to simulate two different CE tubesheet manufacturing techniques. Smooth bore holes represented CE tubesheets fabricated with a bore trepanning association process (such as those at PNP) and rough boreholes represented CE tubesheets fabricated with a gun-drilled process. The tubes were expanded into the full length of the simulated tubesheets (collars) using the standard CE explosive fabrication method. Portions of the tubes were then removed using electrical discharge machining (EDM) to produce simulated tubesheet engagement lengths ranging from 1 inch to 5.5 inches. Leak rates through a 360 degree EDM-generated tube flaw is expected to be greater than leakage from a service induced, through-wall, stress-corrosion crack. Multiple leak tests were performed on each sample to provide data at various tube-to-tubesheet joint lengths and at different test temperatures.

Given the laboratory leak rate data obtained at elevated temperatures (i.e., 600 °F), the licensee's leakage methodology calculates the necessary C* inspection length using several relationships developed in WCAP-16208. The inspection length (uncorrected joint length) required to not exceed 0.2 gpm SG leakage (0.1 gpm for each side of the SG) was determined using the relationship between tube-to-tubesheet joint length and leak rate developed from the leak rate tests. This inspection length was then analytically corrected assuming the limiting MSLB conditions, when tubesheet bow and accompanying tubesheet hole dilation effects are at a maximum. A final (corrected) inspection length was established that accounted for tubesheet hole dilation effects and uncertainty related to NDE probe axial position. This methodology uses the load at first slip (rather than the maximum load or load at first move).

The NRC staff considered the effect of the postulated C* leakage on the margin between accident-induced leakage and operational leakage. Since the PNP accident-induced leakage limit is 0.3 gpm per SG, and the C* methodology assumes accident-induced leakage of 0.2 gpm per SG, the leakage from all sources other than C* implementation can be no more than 0.1 gpm per SG. The TS operational leakage limit is 150 gallons per day (0.1 gpm) through any one SG. Since an operational leakage source may leak at a higher rate under accident conditions than under normal operating conditions, it may be necessary to keep the observed operational leakage below the operational leakage limit to ensure the accident-induced leakage limit is not exceeded during an accident. As discussed in NRC Regulatory Information Summary 2007-20, "Implementation of Primary-to-Secondary Leakage Performance Criteria" (ADAMS Accession No. ML070570297), the licensee may have to implement more restrictive operational leakage limits to ensure the accident-induced leakage limit is not exceeded during an accident.

During the review of the licensee's leakage methodology, the NRC staff noted some inconsistencies in the leak rate data and potential uncertainties introduced into the data.

These include:

- The leak rate response to increasing differential pressure was not consistent between the room temperature leak rate tests at the Westinghouse Windsor facility and the Westinghouse Science and Technology Division (STD) facility. For example, in some cases, the leak rate increased with increasing differential pressure and in other cases, the leak rate was constant or decreased with increasing differential pressure. The leak rates in the STD tests appeared to increase consistently with increasing differential pressure.

- Although earlier testing in support of the W* methodology showed leak rates decreased as temperature increased (from room temperature to operating temperatures), operating temperature leak rates in the C* tests were greater than the corresponding room temperature leak rates in many cases. Similarly, the W* tests indicated the leak rate was relatively independent of the differential pressure and the C* tests indicated that the leak rate increases with differential pressure. (The W* methodology is a similar methodology to C*, but is applied to SGs with explosively expanded tube-to-tubesheet joints that are in Nuclear Steam Supply Systems (NSSS) designed by Westinghouse, while C* is applied to SGs with explosively expanded tube-to-tubesheet joints that are in NSSS designed by Combustion Engineering.)

- Leak tests in support of the C* methodology were conducted at two facilities with different test techniques and different leak rate measurement techniques. Since initial tests of samples at the STD facilities yielded leak rates that were significantly less than the last comparable leak rates measured at the Windsor facility, an all volatile water treatment (AVT) was applied to some of the test samples. Leak rate tests performed after the AVT treatment resulted in an increase in the leak rate, suggesting oxides developed in and partially blocked the tube-to-collar crevices either during earlier tests or during post-test handling at the first facility (Windsor). No destructive examination was performed to characterize these crevices during testing or at the completion of testing. Almost all of the room temperature leak rates measured at the Windsor facility are greater than the leak rate measured at STD (for the same specimen).

- The leak rate was higher in the heatup phase than in the cooldown phase for some specimens, while the opposite trend occurred in other specimens.

- The leak rate data were determined to be independent of the tubesheet hole roughness (rough bore or smooth bore) whereas the resistance to tube pullout is dependent on the tubesheet hole roughness (smoother bore holes are less resistant to tube pullout).

- Multiple tests were performed on the same specimen. These multiple tests included both room temperature and elevated temperature tests along with tests of various crevice lengths. The initial tests may have introduced deposits into the crevice, which could have restricted the leak rate in subsequent tests.

- The leak rate decreased with time for the C* tests. This trend was not always observed in similar tests performed for tubes hydraulically expanded into a tubesheet collar (i.e., H* tests).

- The determination that the EDM process had no effect on obstructing the leak path was based on one specimen (albeit at several locations within that specimen).

- The tube and collar temperatures were not monitored during the welding and cutting of the specimens, introducing uncertainty on whether the joint loosened or whether oxides could have formed in the crevice.

- The surface finish of the specimens was not measured, thereby introducing uncertainty about whether the surface finish of the specimens is comparable to that in the field.

In addition to the above, the NRC staff notes that the inspection distance associated with leakage was determined from the correlation of joint length to the load at first slip, rather than from a correlation of joint length to the load at first move, and the test data indicate the leak rate at some lower temperatures (e.g., 460 °F) may be greater than the leak rate at 600 °F. Since the leak rate through a flaw in a tube within the tubesheet is a complex function of several factors, it is reasonable to expect some inconsistencies in the data. These factors include the trapping of corrosion products between the tube and tubesheet, the formation of oxides before and during the occurrence of leakage, the deposition of boric acid in the "crevice" after leakage initiates, viscosity of the fluid, contact pressure, the tube and the tubesheet's response to changing temperature conditions (e.g., tube cooling quicker than tubesheet), tubesheet hole asperities, and extrusion of the tube into the asperities during the initial explansion process. As a result, even though the staff noted some inconsistencies and potential uncertainties in the leak rate data, the staff considers the leakage methodology acceptable for the following reasons:

- The licensee will perform inspections and plug all tubes with flaws within 12.5 inches below the top of the hot-leg tubesheet (or BET, whichever is lower) and within 13.67 inches below the top of the cold-leg tubesheet (or BET, whichever is lower).

- The model used for calculating dilation of the tubesheet holes is based on the most dilated hole in the tubesheet. No credit is taken for the significant reduction in total leakage that would be realized by applying less dilation to the other radial positions of the tubesheet, such as near the periphery and near the stay cylinder.

- The licensee assumes all tubes remaining in service contain a 360-degree circumferential, 100 percent through-wall flaw (i.e., a tube sever) at the bottom of the C* distance. This assumption is conservative given industry inspection results within the tubesheet region.

- EDM slits used to simulate circumferential cracks for the leak rate tests are wider and restrict flow less than service-related stress corrosion cracks. In addition, the accumulation of sludge and corrosion products at the secondary face of the PNP SG tube-to-tubesheet joint is expected to restrict flow more than the leak test samples.

- Flaws postulated below the C* distance are assumed to be leaking although industry operating experience has demonstrated negligible leakage under normal operating conditions, even when cracks are located in the expansion transition zone near the top of the tubesheet.

- No credit is taken for corrosion in the tubesheet joint, which would be expected to at least partially block the leak path and significantly reduce the total leak rate.

In summary, the NRC staff concludes that the proposed C* distance of 13.67 inches below the top of the cold-leg side of the tubesheet is acceptable to ensure that the amount of accident-induced leakage from undetected flaws below the C* distance (i.e., the inspection distance) will be limited to less than the accident-induced leakage limit.

3.5 Conclusion

The NRC staff concludes that the licensee's proposed C* methodology for assessing structural and leakage integrity for flaws in the cold-leg side of the SG tubesheets is acceptable. Therefore, the staff concludes that the licensee's proposal to limit the extent of tube inspections in the cold-leg side of the SG tubesheet and to plug all tubes with service-induced flaws found within the C* distance, is acceptable and the proposed changes are in accordance with 10 CFR 50.36.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, or any effluents that may be released offsite, and that there is no significant increase in individual, or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (81 FR 50747, August 2, 2016). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

Principal Contributor: A. Johnson

Date of issuance: December 19, 2016

A copy of our related safety evaluation is provided in Enclosure 2. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Jennivine K. Rankin, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 261 to DPR-20
2. Safety Evaluation

cc w/encls: Distribution via Listserv

DISTRIBUTION
PUBLIC
LPL3-1 r/f
RidsACRS_MailCTR Resource
RidsNrrDorlLpl3-1 Resource
RidsNrrDssSrxb Resource
RidsNrrDssStsb Resource
RidsNrrLASRohrer Resource
RidsNrrPMPalisades Resource
RidsRgn3MailCenter Resource
AJohnson, NRR/DE

ADAMS Accession No: ML16300A030 ***via memo**

OFFICE	NRR/DORL/LPL3-1/PM	NRR/DORL/LPL3-1/LA	NRR/DE/ESGB/BC	NRR/DSS/STSB/BC
NAME	JRankin	SRohrer	JPeralta (A)*	AKlein
DATE	10/27/16	10/27/16	9/22/16	11/2/16
OFFICE	OGC-NLO Subject to edits	NRR/DORL/LPL3-1/BC	NRR/DORL/LPL3-1/PM	
NAME	BHarris	DWrona	JRankin	
DATE	11/7/16	12/19/16	12/19/16	

OFFICIAL RECORD COPY