May 31, 2007

Mr. Michael Balduzzi
Sr. Vice President, Regional Operations NE
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT RE: TUBESHEET INSPECTION DEPTH FOR STEAM GENERATOR TUBE INSPECTIONS (TAC NO. MD2125)

Dear Mr. Balduzzi:

The Commission has issued the enclosed Amendment No. 225 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant (PNP). The amendment consists of changes to the Technical Specifications (TSs) in response to the application dated May 30, 2006, as supplemented by letters dated February 27, and April 10, 2007.

The amendment revises the Palisades Technical Specification (TS) Section 5.5.8. The change revises the repair criteria and essentially results in the licensee's not having to inspect the lower portion of the tube within the tubesheet (since all flaws in this region are acceptable).

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Registe*r notice.

Sincerely,

/RA/

Mahesh L. Chawla, 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. 225 to DPR-20
2. Safety Evaluation

cc w/encls: See next page

May 31, 2007

Mr. Michael Balduzzi
Sr. Vice President, Regional Operations NE
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: PALISADES NUCLEAR PLANT - ISSUANCE OF AMENDMENT RE: TUBESHEET INSPECTION DEPTH FOR STEAM GENERATOR TUBE INSPECTIONS (TAC NO. MD2125)

Dear Mr. Balduzzi:

The Commission has issued the enclosed Amendment No. 225 to Renewed Facility Operating License No. DPR-20 for the Palisades Nuclear Plant (PNP). The amendment consists of changes to the Technical Specifications (TSs) in response to the application dated May 30, 2006, as supplemented by letters dated February 27, and April 10, 2007.

The amendment revises the Palisades Technical Specification (TS) Section 5.5.8. The change revises the repair criteria and essentially results in the licensee’s not having to inspect the lower portion of the tube within the tubesheet (since all flaws in this region are acceptable).

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Registe*r notice.

Sincerely,

/RA/

Mahesh L. Chawla, 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. 225 to DPR-20
2. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION

PUBLIC	LPL3-1 r/f	RidsNrrDorlLpl3-1
RidsNrrPMMChawla	RidsNrrLATHarris	RidsOGCRp
RidsAcrsAcnwMailCenter	RidsNrrDirsItsb	G. Hill, OIS
RidsRgn3MailCenter	RidsNrrDorlDpr	G. Makar
A. Hiser		

ADAMS ACCESSION NO: ML071420216, Pkg: ML071420240, TS: ML071450105 *with comments

OFFICE	LPL3-1/PM	LPL3-1/LA	TECH BR	OGC	LPL3-1/BC
NAME	MChawla	THarris	AHiser	AHodgdon (NLO)*	LRaghavan
DATE	05/15/07	05/15/07	05/28/07	05/25/07	05/ 31 /07

OFFICIAL RECORD COPY

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Suite 210
2443 Warrenville Road
Lisle, IL 60532-4351

Supervisor
Covert Township
P. O. Box 35
Covert, MI 49043

Office of the Governor
P. O. Box 30013
Lansing, MI 48909

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Palisades Plant
27782 Blue Star Memorial Highway
Covert, MI 49043

Michigan Department of Environmental Quality
Waste and Hazardous Materials Division
Hazardous Waste and Radiological
Protection Section
Nuclear Facilities Unit
Constitution Hall, Lower-Level North
525 West Allegan Street
P.O. Box 30241
Lansing, MI 48909-7741

Michigan Department of Attorney General
Special Litigation Division
525 West Ottawa St.
Sixth Floor, G. Mennen Williams Building
Lansing, MI 48913

Mr. Michael R. Kansler
President & CEO/CNO
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John T. Herron
Sr. Vice President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Sr. Vice President,
Engineering and Technical Services
Entergy Nuclear Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213

Mr. Bruce C. Williams
Vice President, Oversight
Entergy Nuclear Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213

Mr. Christopher J. Schwarz
Site Vice President
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

General Manager, Plant Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

Mr. Oscar Limpias
Vice President, Engineering
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John F. McCann
Director, Nuclear Safety & Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue

White Plains, NY 10601

Ms. Charlene D. Faison
Manager, Licensing
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Ernest J. Harkness
Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. William Dennis
Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Joseph DeRoy
Vice President, Operations Support
Entergy Nuclear Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213

Laurie A. Lahti, Manager, Licensing
Regulatory Affairs
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

ENTERGY NUCLEAR OPERATION, INC.

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 225
Renewed License No. DPR-20

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

   A. The application for amendment by Nuclear Management Company, LLC (the licensee, at the time of submittal), dated May 30, 2006, supplemented by letters dated February 27, and April 10, 2007, 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;

   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. 225 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Renewed Facility Operating License
and Technical Specifications

Date of Issuance: May 31, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 225

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 changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

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

REMOVE	INSERT
5.0-12	5.0-12
5.0-13	5.0-13

(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. 225 , 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) ENO shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SERs dated 09/01/78, 03/19/80, 02/10/81, 05/26/83, 07/12/85, 01/29/86, 12/03/87, and 05/19/89 and subject to the following provisions:

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 225 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-20

ENTERGY NUCLEAR OPERATION, INC.

PALISADES NUCLEAR PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated May 30, 2006 (ML061560406), as supplemented by letters dated February 27, (ML070640056), and April 10, 2007 (ML071030330), Nuclear Management Company (NMC, the licensee, at the time of submittal) requested changes to the Technical Specifications (TSs) for Palisades Nuclear Plant (PNP). The proposal would revise the requirements for steam generator (SG) tube repair in the hot-leg tubesheet region by applying a methodology called C* (C-star). Entergy Nuclear Operations, Inc. has since become the current licensee, following a license transfer that occurred on April 11, 2007.

The C* methodology was developed for Combustion Engineering (CE) plants with tubes that were expanded into the tubesheet with an explosive process called "explansion." The existing PNP TS 5.5.8.c specifies that tubes containing flaws equal to or greater than 40 percent in depth must be plugged. The proposed amendment modifies TS 5.5.8.c to specify that the 40 percent depth criteria for tube repair does not need to be applied in the hot-leg tubesheet region below a certain elevation, generally referred to as the C* distance. According to the C* methodology in TS 5.5.8.c, flaws below this elevation may remain in service regardless of size.

Implementing the C* methodology also eliminates the need to inspect the portion of the tube within the hot-leg tubesheet region below the C* distance, since the inspection provision in TS 5.5.8.d 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. Flaws located in the hot-leg region below the top of the tubesheet (TTS) or bottom of the expansion transition (BET), whichever is higher, but above the C* elevation proposed in TS 5.5.8.c, would have to be plugged.

The supplements provided additional information that clarified the application, but did not expand the scope of the application as originally noticed and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 24, 2006 (71 FR 62310).

2.0 REGULATORY EVALUATION

SG tubes function as an integral part of the reactor coolant pressure boundary and, in addition, 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 characterizing the nature and cause of any tube flaws so that corrective measures can be taken. Tubes with flaws that exceed the tube repair limits specified in a plant's TS are removed from service by plugging or are repaired by sleeving (if approved by the NRC for use at the plant). 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 TSs. 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 measures ensure that the radiological dose consequences associated with any leakage are within acceptable limits and that they limit the frequency of SG tube ruptures.

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 Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A General Design Criteria (e.g., GDC 14, 32) are satisfied. The NRC staff's evaluation is based, in part, on ensuring that the structural margins inherent in Regulatory Guide 1.121, "Bases for Plugging Degraded PWR SG Tubes," are maintained. The NRC staff's evaluation also includes verifying that a conservative methodology exists for determining the amount of primary-to-secondary leakage that may occur during design-basis accidents. 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, GDC 19 of Appendix A to 10 CFR Part 50, and 10 CFR 50.67.

In 2006, the NRC approved similar redefinitions of the tube repair criteria for St. Lucie Unit 2 (ML060790352), Waterford Unit 3 (ML062420419), and San Onofre Units 2 and 3 (ML062970441). In each case, plant-specific repair criteria were determined.

3.0 TECHNICAL EVALUATION

3.1 Background

PNP is a 2-loop, CE design plant with Model 2530 replacement SGs. Each SG contains approximately 8219 mill-annealed Alloy 600 tubes with an outside diameter of 0.75 of an inch and a wall thickness of 0.042 of an inch. The tubes were explosively expanded ("explansion" process) at both ends for the full length of the tubesheet. The tubesheets were drilled using a bore trepanning process and the resulting surface finish is termed "smooth bore." A weld joins each tube end to the cladding on the primary face of the tubesheet, providing a leak-tight boundary and resistance to tube pullout. The explansion process produces an interference fit between the tube and tubesheet, which can also provide resistance to tube pullout. The transition from the expanded portion of the tube to the unexpanded portion of the tube is referred to as the expansion transition.

Each SG contains horizontal supports (eggcrates), diagonal supports (batwings), and vertical supports. All tube support material is carbon steel. The shorter tubes, located in the first 18 rows at the center of the tube bundle, have 180-degree bends (i.e., are bent in the shape of a "U"). Higher row number tubes (Rows 19-165) have double 90-degree bends with a horizontal run between the bends ("square bends").

The tube-to-tubesheet joint consists of the tube, which is explosively expanded against the bore of the tubesheet, the tube-to-tubesheet weld located at the tube end, and the tubesheet. Typically, plants designed the tube-to-tubesheet joint as a welded joint rather than a friction or expansion joint. That is, the weld itself was designed as a pressure boundary element, and it was designed to transmit the entire end cap pressure load during normal and design-basis accident conditions from the tube to the tubesheet with no credit taken for the friction developed between the explosively expanded tube and the tubesheet. In addition, the weld serves to make the joint leak tight.

The existing inspection requirements in the plant TSs 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 essentially precluding 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 defect.

Based on these considerations, power reactor licensees have proposed, and the NRC has approved, alternate repair criteria for defects located in the SG tube contained in the lower portion of the tubesheet, when these defects are a specific distance below the expansion transition or the top of the tubesheet, 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) is 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 undegraded tubing needed to ensure the tube will not pull out of the tubesheet and by determining that the amount of leakage from flaws below the C* distance is limited (i.e., within acceptance limits). The C* distance is measured from the top of the tubesheet or the bottom of the expansion transition, 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 bottom of the expansion transition and the total inspection distance. They are based on a 95-percent uncertainty adjustment of previous data for a similar tubesheet inspection methodology for Westinghouse designed plants (W*) and were reviewed and approved as part of the NRC staff's safety evaluation approving the W* repair criteria for Diablo Canyon Units 1 and 2.

The C* analysis presented in Westinghouse Commercial Atomic Power report, WCAP-16208, Revision 1, uses non-plant-specific values for primary system pressure, secondary pressure, and primary temperature to determine the C* distance for tubing within the tubesheet. The C* distance was adjusted for the hot-leg temperature at PNP, since the hot-leg temperature at the plant is lower (less conservative) than that assumed in the generic C* analysis.

The C* analysis considered the forces acting to pull the tube out of the tubesheet (i.e., from the internal pressure in 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.

3.2 Palisades's Proposal

The licensee's basis for revising the criteria for tube repair within the tubesheet regions is documented in its license amendment request; in Westinghouse LTR-CDME-06-80, Revision 1, "Palisades Tubesheet Inspection Depth"; in Westinghouse LTR-CDME-06-40, Revision 1, "Comments on the Application of WCAP-16208-P, Revision 1, 'NDE Inspection Length for CE Steam Generator Tubesheet Region Explosive Expansions,' to the Palisades Nuclear Power Plant"; and in its supplemental letters listed above. WCAP-16208, Revision 1, describes the analysis and testing performed to justify a similar modification in the tube repair criteria for St. Lucie Unit 2.

Proposed TS 5.5.8.c specifies that tubes found by inservice inspection to contain flaws with a depth equal to or greater than 40 percent of the nominal tube wall thickness shall be plugged. An alternative to the repair criteria is proposed in TS 5.5.8.c.1, which specifies a tube must be plugged if any flaw is detected within 12.5 inches below the bottom of the hot-leg expansion transition or top of the tubesheet, whichever is lower. The alternate repair criteria (i.e., C* criteria) proposed in TS 5.5.8.c.1 allow tubes with flaws to remain in service if the flaws are located greater than 12.5 inches below the hot-leg TTS or BET, whichever elevation is lower.

The following sections summarize the NRC staff's evaluation of the licensee's C* proposal for PNP in terms of maintaining SG tube structural and leakage integrity.

3.3 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 of time between inspections. Tube rupture and the tube pullout from the tubesheet are the two potential modes of structural failure considered for tubes returned to service under the C* methodology.

In order for a tube to rupture as a result of implementing the C* criteria, a flaw would need to grow above the tubesheet's secondary face. If the entire flaw remains within the tubesheet, the reinforcement provided by the tubesheet will prevent tube rupture. The C* methodology proposed by Palisades requires an inspection of the C* distance and the repair of any flaws in the C* distance. Therefore, any known flaws remaining in service following the examinations will be located a minimum of 12.5 inches below the top of the hot-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 12.5 inches below the top of the tubesheet to outside the tubesheet. Therefore, it is unlikely that any of these flaws will grow in an axial direction and extend outside the tubesheet during one operating cycle. Thus, tube rupture is precluded for these flaws due to the reinforcement provided by the surrounding tubesheet.

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. Differential pressures from the primary side to the secondary side of the SG impart axial loads into each tube that are reacted at the tube-to-tubesheet interface. Axial tube loading during normal operating conditions can be significant. The peak postulated loading, however, occurs during events involving a depressurization of the secondary side of the SG (e.g., main steamline 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 (i.e., explansion) process. In addition, the differential thermal expansion between the tube and the tubesheet and the internal pressure of the tube tightens 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.

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.

To justify the acceptability of any type or combination of tube flaws below the C* distance from a structural integrity standpoint, the licensee completed an assessment using analytical calculations and laboratory experiments. The laboratory experiments included tests of prototypical SG tube-to-tubesheet joints to evaluate the length of sound tubing necessary to maintain structural integrity (i.e., to ensure the tube would not pull out of the tubesheet under axial loading conditions). Laboratory data from two sources were used in performing this assessment. These data include data obtained under Combustion Engineering Owners Group (CEOG) Task 1154 and data obtained during the development of WCAP-16208. These latter data are referred to as pullout screening tests since they were performed simply to confirm that the joint length would satisfy the limiting load condition (i.e., three times the normal operating load) rather than determining the axial load resistance provided by the joint. On the other hand, the Task 1154 samples were tested to determine the axial load resistance of various joint lengths. The Task 1154 data are from the following sources:

- An 8-inch thick, seven tube-to-tubesheet joint mockup built for Ringhals
- A portion of the as-built SG tubesheet from a cancelled Boston Edison contract
- Tube-to-tubesheet mockups fabricated for CEOG Task 1154 tube pullout and leakage tests

Results from pullout testing of the Task 1154 prototypical SG tube-to-tubesheet joints provided the basis for determining the length of sound tubing necessary to maintain the appropriate structural margins for tubes that may be severed from flaws within the tubesheet. The test specimens were subjected to axial loadings in order to demonstrate acceptable structural capabilities. The axial resistance provided by these specimens was evaluated using the load required to move the tube within the tubesheet (maximum load). To determine the axial load resistance provided by just the explansion process, only the tests performed under ambient pressure and room temperature conditions were evaluated. Adjustments were made to the data to account for differences in testing and in the tube wall thickness. A 95-percent lower bound fit to the adjusted data was used in evaluating the axial load resistance provided by the explansion process. The pullout screening tests discussed above confirmed the adequacy of the pullout forces measured under the Task 1154 program.

In addition to the resistance from pullout provided by the explansion process, the following were also considered in determining the appropriate inspection distance from a structural integrity (tube pullout) standpoint: (1) resistance to tube pullout from the differential thermal expansion between the tube and the tubesheet, (2) resistance to tube pullout from the internal pressure in the tube within the tubesheet, (3) uncertainty in determining the amount of tube inspected, and (4) reduction in the resistance to tube pullout from the bowing of the tubesheet.

Consideration of the effects of pressure and temperature were based on analytical calculations (rather than test results). Although the increase in joint tightness (i.e., increase in the axial load resistance of the tube-to-tubesheet joint) as a result of internal pressure and differential thermal expansion between the tube and tubesheet was addressed analytically, the pullout tests performed at operating pressure and/or temperature were reviewed to verify that the results from these tests supported the analytical adjustments made to account for these effects.

Consideration of the uncertainty in determining the amount of tube inspected (i.e., the NDE axial position uncertainty) was based on testing as discussed in WCAP-16208. The uncertainties used in determining the appropriate C* inspection distance were based on data obtained during the development of the W* inspection distance since both involve inspection of tubes explosively expanded into the tubesheet.

Consideration of the effects of tubesheet bow were based on analytical (finite element) calculations as discussed in WCAP-16208. Since the analytical model considered the point of maximum tubesheet deflection, it will result in the most dilation and the greatest reduction in the axial load carrying capability of the tube-to-tubesheet joint.

The above factors were combined and it was determined that an inspection distance of approximately 5.3 inches was required to prevent tube pullout, which is significantly less than the C* inspection distance of 12.5 inches proposed by the licensee. This is because the C* inspection distance is governed by leakage integrity considerations rather than structural integrity considerations.

With respect to the licensee's assessment of tube pullout, the NRC staff notes that instead of considering a lower 95-percent bound of the tube pullout data, it considered the most limiting pullout data since all tubes should resist pullout from the tubesheet. In addition, the NRC staff considered the load associated with the first movement of the joint, since movement of this joint was not considered (to the NRC staff's knowledge) in the original design of the tube-to-tubesheet joint. This consideration would only increase the inspection distance for ensuring structural integrity by a minor amount. As a result, the structural integrity inspection distance is still significantly less than the proposed inspection distance of 12.5 inches and is considered acceptable. In addition, the NRC staff notes that it did not review the scaling procedures that Palisades applied to the tube pullout data in detail; however, the adjustments made appeared reasonable and even if no adjustments were made to the data it would not have significantly affected the inspection distance.

With respect to the finite element model, the NRC staff reviewed the input assumptions and the results of the model. As a result of the NRC staff's review, the NRC staff concludes that the results of the analysis are comparable to other analyses reviewed and approved by the NRC staff given the design configuration of the CE SGs (e.g., stay cylinder, explosive expansions).

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, which precludes tube burst; past inspection results, which indicate that 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 (e.g., worst-case tube dilation, use of limiting pullout data, use of a 95-percent NDE uncertainty adjustment), the confirmation through testing that the analytical adjustments for pressure and temperature are supported; and the restriction to tube pullout provided 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 takes no credit for restriction to tube movement inherent to the SG design.

3.4 Tube Leakage Integrity

In assessing leakage integrity of an 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. The licensee reported this plant-specific limit for PNP is 0.3 gallons per minute (gpm) total through both SGs. 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 0.1 gpm per SG under the most limiting design-basis accident conditions (MSLB). That is, the inspection distance required by this proposal (12.5 inches) was determined based on ensuring the leakage from flaws within the tubesheet region would contribute no more than 0.1 gpm to the total leakage under MSLB conditions. The 0.1 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 total for the two SGs.

The licensee's method for determining the amount of leakage from flaws within the tubesheet region depends on whether the flaws are located within the C* distance or below the C* distance. For flaws located within the C* distance, no leakage is anticipated since the proposed TSs state that all flaws in this region will be plugged. As a result, the only flaws expected within the C* distance would either be 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 accident conditions. Although no leakage is expected from flaws within the C* distance, the licensee committed to assess potential leakage from such flaws as part of its assessments.

Since the C* methodology does not require inspections below the C* distance, there is a potential that flaws that 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 the hot-leg region. The assumed number of 7846 tubes per SG was selected based on the SG with the greatest number of inservice tubes, and is therefore conservative with respect to leakage. Given past plant-specific and industry operating experience, the NRC staff considers it conservative to assume all tubes contain circumferential, through-wall flaws at the C* distance.

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 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 carbon steel collar hole finishes were tested to simulate two different CE tubesheet manufacturing techniques. The drilled and reamed collar hole specimens were used to represent CE tubesheets fabricated with a bore trepanning association process (smooth bore), such as those at PNP, and the drilled collar hole specimens were used to represent CE gun-drilled (rough bore) tubesheets. 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 electric discharge machining (EDM) to produce tube-to-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 0.1 gpm SG leakage 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 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, and the C* methodology assumes accident-induced leakage of 0.1 gpm, the leakage from all sources other than C* implementation can be no more than 0.2 gpm. Since the operational (TS) leakage limit is 0.1 gpm per SG, the accident-induced leakage limit is not exceeded when the postulated C* leakage source is added to the operational leakage limit. However, 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.

During its 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 consistently increase 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.

- 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 (the smoother bore tubesheet 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, 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 (1) 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 (2) 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 NRC staff noted some inconsistencies and potential uncertainties in the leak rate data, the NRC staff considers the leakage methodology acceptable for the following reasons:

- The licensee will perform inspections and repair all service-induced degradation to a minimum depth below the top of the hot-leg tubesheet (or expansion transition) of 12.5 inches.

- The tubesheet hole dilation model 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 tube-to-tubesheet joint length (or inspection distance) 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 a small fraction of the accident-induced leakage limit.

3.5 Reporting Requirements

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. 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. These reports will permit the NRC staff to verify the operating experience continues to be conservative relative to the assumptions made in the amendment. As a result, the NRC staff concludes the TS reporting requirements are adequate for ensuring the assumptions used as the basis for approval remain valid.

4.0 SUMMARY

The NRC staff concludes Palisades's proposed methodology for assessing structural and leakage integrity for flaws in the tubesheet region is acceptable. Therefore, the NRC staff concludes that Palisades's proposal to limit the extent of tube inspections in the hot-leg tubesheet regions, and to repair all degradation detected in the region of the tubesheet required to be inspected is acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of 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 (71 FR 62310). 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 need 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) 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: G. Makar

Date: May 31, 2007