UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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December 18, 2018

Mr. Charles Arnone
Vice President, Operations
Entergy Nuclear Operations, Inc.
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043–9530

SUBJECT: REISSUE—PALISADES NUCLEAR PLANT—DESIGN BASES ASSURANCE INSPECTION (TEAMS) INSPECTION REPORT 05000255/2017007

Dear Mr. Arnone:

On November 17, 2017, the U.S. Nuclear Regulatory Commission (NRC) completed Design Basis Assurance Inspection (Teams) at your Palisades Nuclear Plant. On December 29, 2017, the NRC issued Inspection Report 05000255/2017007. This original inspection report documented two NRC-identified findings of very-low safety significance with associated violations of NRC requirements. These violations were treated as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy. In a letter dated November 28, 2018 (ML18337A434), the NRC withdrew the NCV associated with Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion XI, “Test Control,” based on an NRC independent review. The purpose of this letter is to re-issue the NRC Inspection Report 05000255/2017007 in its entirety.

The inspection examined activities conducted under your license as they relate to safety and compliance with the Commission’s rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of this inspection, the NRC has identified one issue that was evaluated under the risk significance determination process as having very-low safety significance (Green). The NRC has also determined that one violation is associated with this issue. Because the licensee initiated condition reports to address this issue, this violation is being treated as a NCV, consistent with Section 2.3.2 of the Enforcement Policy. The NCV is described in the subject inspection report.

This letter, its enclosure, and your response (if any) will be made available for public inspection and copying at http://www.nrc.gov/reading-rm/adams.html and at the NRC Public Document Room in accordance with Title 10 of the *Code of Federal Regulations*, Part 2.390, "Public Inspections, Exemptions, Requests for Withholding."

Sincerely,

/RA/

Kenneth G. O'Brien, Director
Division of Reactor Safety

Docket No. 50–255
License No. DPR–20

Enclosure:
IR 05000255/2017007

cc: Distribution via LISTSERV®

Letter to Charles Arnone from Kenneth G. O'Brien dated December 18, 2018.

SUBJECT: REISSUE—PALISADES NUCLEAR PLANT—DESIGN BASES ASSURANCE INSPECTION (TEAMS) INSPECTION REPORT 05000255/2017007

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–255
License No:	DPR–20
Report No:	05000255/2017007
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	October 30, 2017, through November 17, 2017
Inspectors:	J. Benjamin, Senior Reactor Inspector, Lead B. Jose, Senior Reactor Inspector, Electrical J. Bozga, Senior Reactor Inspector, Structural J. Robbins, Reactor Inspector, Operations G. Nicely, Electrical Contractor J. Zudan, Mechanical Contractor
Approved by:	K. O'Brien, Director Division of Reactor Safety

Enclosure

SUMMARY

Inspection Report 05000255/2017007, 10/30/2017–11/17/2017; Palisades Nuclear Plant; Design Bases Assurance Inspection (Teams).

The inspection was a 2-week onsite baseline inspection that focused on the design of components and modifications to mitigating systems. The inspection was conducted by regional engineering inspectors and two consultants. The inspection team identified a finding of very-low safety significance (Green) associated with one violation of U.S. Nuclear Regulatory Commission (NRC) regulations. The significance of inspection findings is indicated by their color (i.e., Green, White, Yellow, and Red) and determined using Inspection Manual Chapter 0609, "Significance Determination Process," dated April 29, 2015. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated November 1, 2016. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 6, dated July 2016.

NRC-Identified and Self-Revealed Findings

Cornerstone: Barrier Integrity

Green. The inspectors identified a finding of very-low safety significance (Green) and an associated Non-Cited Violation of Title 10 of the *Code of Federal Regulations*, Part 50, Appendix B, Criterion III, "Design Control," for failure to meet Updated Final Safety Analysis Report requirements for containment spray piping supports, specifically straps. Specifically, the inspectors identified that Calculation No. EA-SP-03369-02, Revision 0, used inelastic acceptance limits for the pipe straps which connect the pipe to the pipe support, in order to demonstrate Class I compliance which was not in accordance with the design and licensing basis specification. The license entered the issue into their Corrective Action Program as CR-PLP-2017-05246, "Spray Pipe Support," dated November 14, 2017. The licensee performed an analysis to establish reasonable assurance of operability and the inspectors with support from the Office from the Nuclear Reactor Regulation reviewed this operability and no performance deficiencies were identified.

The performance deficiency was determined to be more-than-minor because it was associated with the Barrier Integrity Cornerstone attribute of design control and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. This finding is of very-low safety significance (Green) because there was no actual reactor containment barrier degradation. The inspectors did not identify a cross-cutting aspect associated with this finding because this was a legacy design issue; and therefore, was not reflective of current performance. (Section 1R21.5.b(1))

REPORT DETAILS

(1) REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Design Bases Assurance Inspection (Teams) (71111.21M)

.1 Introduction

The objective of the design bases assurance inspection is to verify that design bases have been correctly implemented for the selected risk significant components and modifications, and that operating procedures and operator actions are consistent with design and licensing bases. As plants age, their design bases may be difficult to determine and an important design feature may be altered or disabled during a modification. The inspection also monitors the implementation of modifications to structures, systems, and components as modifications to one system may also affect the design bases and functioning of interfacing systems as well as introduce the potential for common cause failures. The Probabilistic Risk Assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems, and Barrier Integrity cornerstones for which there are no indicators to measure performance.

Specific documents reviewed during the inspection are listed in the Attachment to the report.

.2 Inspection Sample Selection Process

The team initially selected samples that were more safety-significant for shutdown conditions based upon the company's public announcement to permanently shutdown in the fall of 2018. Components related to the spent fuel pool cooling water system, spent fuel pool reactivity control components, and control room emergency ventilation systems were therefore initially chosen.

The licensee made the decision to not shutdown in 2018, but rather continue operations until the spring of 2022 several weeks before the inspection was scheduled to start. Based upon this information, the team selected additional, more traditional, risk significant components based upon the risk insights of the licensee's Probabilistic Risk Assessment model, and the Palisades Nuclear Standardized Plant Analysis Risk model with the assistance of a U.S. Nuclear Regulatory Commission (NRC) Region III senior risk analyst. The team selected both the loss of off-site power (LOOP) and steam generator tube rupture events to refine the final component selection, including the large early release frequency (LERF) component selection.

The team also used additional component information such as a margin assessment in the selection process. This design margin assessment considered original design reductions caused by design modifications, power uprates, or reductions due to degraded material condition. Equipment reliability issues were also considered in the selection of components for detailed review. These included items such as performance test results, significant corrective actions, repeated maintenance activities, Maintenance

Rule (a)(1) status, components requiring an operability evaluation, system health reports, and NRC resident inspector input of problem areas/equipment. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins. A summary of the reviews performed and the specific inspection findings identified are included in the following sections of the report.

The team also identified modifications for review with a focus on modifications implemented within the last 3 years. In addition, the inspectors selected procedures and operating experience issues associated with the selected components and other risk informed factors.

This inspection constituted 21 samples (10 components with 1 component associated with LERF implications, 6 modifications, and 5 operating experience) as defined in Inspection Procedure 71111.21M-02.01. The team applied approximately 80 percent inspection effort on the traditional risk-significant component, operating experience, and modification samples. The remaining approximate 20 percent effort was used to review the original shutdown component selection.

.3 Component Design

a. Inspection Scope

The team reviewed the Updated Final Safety Analysis Report (UFSAR), Technical Specifications, design basis documents, drawings, calculations and other available design basis information, to determine the performance requirements of the selected components. The team used applicable industry standards, such as the American Society of Mechanical Engineers Code, Institute of Electrical and Electronics Engineers Standards, and the National Electric Code, to evaluate acceptability of the systems' design. The NRC also evaluated licensee actions, if any, taken in response to NRC issued operating experience, such as Bulletins, Generic Letters, Regulatory Issue Summaries, and Information Notices. The review was to verify that the selected components would function as designed when required and support proper operation of the associated systems. The attributes that were needed for a component to perform its required function included process medium, energy sources, control systems, operator actions, and heat removal. The attributes to verify that the component condition and tested capability was consistent with the design bases and was appropriate may have included installed configuration, system operation, detailed design, system testing, equipment and environmental qualification, equipment protection, component inputs and outputs, operating experience, and component degradation.

For each of the components selected, the inspectors reviewed the maintenance history, preventive maintenance activities, system health reports, operating experience-related information, vendor manuals, electrical and mechanical drawings, operating procedures, and licensee's Corrective Action Program (CAP) documents. Field walkdowns were conducted for all accessible components selected to assess material condition, including age-related degradation, configuration, potential vulnerability to hazards, and consistency between the as-built condition and the design. In addition, the team interviewed licensee personnel from multiple disciplines such as operations, engineering, and maintenance. Other attributes reviewed are included as part of the scope for each individual component.

The following 10 components (samples), including a component with LERF implications, were reviewed:

- <u>Emergency Diesel Generator (EDG) Fuel Oil Transfer Pump (MDP 18A)</u>: The team reviewed drawings and calculations associated with pump motor cable sizing, voltage drop during degraded voltage conditions, and total current drawn by the motor compared to the ampacity of the feeder cables. The team also, reviewed the maximum short circuit current available at the feeder breaker and verified that the breaker interrupting capacity was not exceeded. The team also, reviewed the vendor manual of the pump motor to compare recommended maintenance versus the licensee's actual maintenance activities. The team reviewed calculations for hydraulic performance, net positive suction head (NPSH), required total developed head, pump vibration analysis, pump run-out conditions, potential for vortex formation and loss of suction at the suction source. Seismic design documentation was reviewed to verify pump design was consistent with limiting seismic design conditions. The team also reviewed diesel fuel oil storage requirements for transient and accident conditions.

- <u>Condensate Storage Tank (CST)</u>: The team reviewed seismic and tornado missile calculations, drawings, and operating procedures associated with the CST. The inspectors assessed the stress analysis for portions of the auxiliary feedwater (AFW) piping that is connected to the CST. The inspectors assessed the tank's volume, capacity, levels, and setpoints with respect to AFW pump suction requirements.

- <u>CST Level Transmitters (LT021/22)</u>: The team reviewed the schematic and instrument loop diagrams of the level transmitters, their setting sheets, power supply requirements, calibration data and heat trace requirements of the instrument tubing. The team reviewed the operation of the level transmitters during freezing conditions concurrent with a loss of offsite power or a station black out, since the heat tracings were powered from nonsafety-related sources. The team reviewed the trip circuitry of the AFW pumps on low suction pressure and verified that the low suction pressure switch settings were adequately coordinated with the AFW pump NPSH requirements.

- <u>Positive Displacement Pump (MDP 55A)</u>: The team reviewed the electrical schematic and wiring drawings of the pump motor, name plate data, and minimum voltage requirements compared to the minimum available voltage during degraded voltage conditions. The team also reviewed the feeder cable size compared to the pump motor ampacity requirements. The team verified that the motor name plate data was correctly translated in the electrical load flow, short circuit and voltage drop calculations. Also, the feeder breaker interrupting capacity was verified to be above the maximum short circuit current available at the breaker and the breaker control components. The team reviewed calculations for hydraulic performance, NPSH, required total developed head and pump run-out conditions. Seismic design documentation was reviewed to verify that the pump design was consistent with limiting seismic design conditions. Test results were reviewed to verify acceptance criteria were met and performance degradation would be identified, taking into account set-point tolerances and instrument inaccuracies.

- 2400VAC Safeguard Transformer: The team reviewed loading calculations to determine whether the capacity of the transformer was adequate to supply worst-case loading. Voltage calculations and operating procedures were reviewed to determine whether transformer taps and administrative controls for switchyard voltage were adequate to assure the availability of offsite power during accident conditions. Procedures for preventive maintenance, inspection, and testing were reviewed to compare maintenance practices against industry and vendor guidance.

- 2400VAC Safeguard Bus: The team reviewed calculations for electrical distribution, system load flow/voltage drop, short-circuit, and electrical protection to verify that bus capacity and voltages remained within the minimum acceptable limits. The protective device settings and circuit breaker ratings were reviewed to ensure adequate selective protection coordination of connected equipment during worst-case short circuit conditions. The team verified that degraded and loss of voltage relays and associated time delays were set in accordance with calculations, and that associated calibration procedures were consistent with calculation assumptions, associated time delays, and set point accuracy calculations. The team evaluated selected portions of the licensee response to NRC Generic Letter 2006-02, "Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power," dated February 1, 2006. The team reviewed the station's interface and coordination with the transmission system operator for plant voltage requirements and notification set points. The team reviewed the 125VDC voltage drop calculations to ensure that the EDG breaker controls would have adequate voltage to operate during a LOOP event. The EDG loading calculations were reviewed to determine whether the capacity of the EDGs was adequate to supply worst case accident loads. The last EDG LOOP/loss-of-coolant accident (LOCA) surveillance tests were reviewed to ensure that the voltage and frequency dips and recovery were within the design limits.

- Train 'A" Main Steam Isolation Valve (CV-0510)—LERF Sample: The inspectors reviewed Inservice Testing (IST) stroke time data and Air Operated Valve Program requirements to ensure valve performance was being appropriately monitored. The inspectors reviewed the main steam piping from steam generator 'A' to the containment penetration pipe stress analysis. The team reviewed open and closing force calculations to assess the main steam isolation valve's capability to function as described in the UFSAR under all bounding conditions. The team reviewed power and control wiring diagrams to assess the control and actuation schemes adequacy. This review constituted one component sample with LERF implications.

- AFW Suction Line Low Pressure Switch (PS 0741): The team reviewed the instrument loop and power supply drawings, pressure switch set points and calibration data. The team also, reviewed the AFW pumps low suction pressure trip circuitry and verified that the settings were adequately coordinated to ensure that the AFW pumps had adequate NPSH under design basis conditions.

- "A" Train Spent Fuel Pool Cooling Water Pump: The team reviewed calculations for required hydraulic performance, NPSH, required total developed head, pump vibration analysis, pump run-out conditions, potential for vortex formation, loss of suction at the suction source and alternate means of fuel pool purification with a floating skimmer. Seismic design documentation was reviewed to verify pump design was consistent with design based seismic conditions.

- "A" Train Control Room Recirculation Ventilation Dampers: The team reviewed calculations for load flow/voltage drop and short-circuit to verify that bus capacity and voltages remained within minimum acceptable limits. The protective device settings and circuit breaker ratings were reviewed to ensure adequate selective protection coordination of connected components during worst-case short circuit conditions.

b. Findings

No findings were identified.

.4 Mitigating System Modifications

a. Inspection Scope

The team reviewed five permanent plant modifications. This review included in-plant walkdowns for accessible portions of the modified structures, systems, and components. The team reviewed the modifications to verify that the design bases, licensing bases, and performance capability of the components had not been degraded through modifications. The modifications were selected based upon risk significance, safety significance, and complexity. The team reviewed the modifications selected to determine if:

- the supporting design and licensing basis documentation was updated;

- the changes were in accordance with the specified design requirements;

- the procedures and training plans affected by the modification have been adequately updated;

- the test documentation as required by the applicable test programs has been updated; and

- post-modification testing adequately verified system operability and/or functionality.

The team also used applicable industry standards to evaluate acceptability of the modifications. The modifications listed below were reviewed as part of this inspection effort:

- Engineering Change (EC) 0000058140; Install Permanent Shielding on Letdown Heat Exchanger;

- EC 0000058141; Install Shielding on Pressurizer Surge Line E-50A Platform;

- EC 0000056644; Supplemental Diesel Generator Fuel Oil Tank to Comply with Michigan Fire Code;

- EC 0000048188; FLEX EC#21 – Turbine Driven AFW System FLEX Upgrades;

- EC 0000055367; Install Larger Size Power Cables between EX-04 (SU1-2) and 2400 VAC Buses 1C and 1D; and

- EC 0000071766; 52-389, Replace Control Transformer on CV-1510 MSIV Bypass Valve.

b. Findings

No findings were identified.

.5 Operating Experience

a. Inspection Scope

The team reviewed five operating experience issues (samples) to ensure that generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection:

- Point Beach Containment Dome Truss License Amendment Request 278, "Risk-Informed Approach To Resolve Construction Truss Design Code Non-Conformances"; March 31, 2017;

- CR-PLP-2014-04976; C and D Batteries Part 21 Separator Misalignment LCR, KCR, and LCY Batteries;

- Non-Cited Violation (NCV) 05000285/2012011-04; "Inadequate Design Basis Documentation";

- NRC Information Notice 2012-06; "Design Vulnerability in Electric Power Systems"; and

- NRC Regulatory Information Summary 2011-012, "Adequacy of Station Electrical Distribution System Voltages".

b. Findings

(1) Containment Spray Pipe Support Strap Deficiencies

Introduction: The inspectors identified a finding of very-low safety significance (Green) and an associated potential NCV of Title 10 of the *Code of Federal Regulation* (CFR), Part 50, Appendix B, Criterion III, "Design Control," for failure to meet UFSAR requirements for containment spray (CS) piping supports, specifically straps. Specifically, the strap (connection between pipe and support) design of CS pipe supports HC44-R884, HC44-R884.1, HC44-R884.2, HC44-R884.3, HC44-R884.4, HC44-R884.5 and HC44-R884.9 did not comply with UFSAR Section 5.10.1.2 and Specification No. C-173(Q) requirements.

Description: The CS system per UFSAR, Section 6.2.1, has the following safety-related design basis functions: The function of the CS system is to limit the containment building pressure rise and reduce the airborne radioactivity in containment by providing a means for spraying the containment atmosphere after occurrence of a LOCA or a main steam line break. The CS piping and pipe supports were designed to Class I requirements as described in UFSAR, Section 5.10.1.2, titled CP Co Design Class 1 Pipe Supports and Specification No. C-173(Q), “Technical Requirements for the Analysis and Design of Safety-Related Pipe Supports,” Revision 6. This specification was classified as safety-related. Calculation No. EA-SP-03369-02, “Containment Spray System Pipe Supports,” Revision 0 evaluated CS pipe supports HC44-R884, HC44-R884.1, HC44-R884.2, HC44-R884.3, HC44-R884.4, HC44-R884.5 and HC44-R884.9 in accordance with Class I requirements for all design basis loading. The pipe supports were analyzed to withstand applied stress due to dead loads, live loads, seismic loads, and thermal loads. The inspectors identified that in Calculation No. EA-SP-03369-02, Revision 0, the licensee used inelastic acceptance limits for the pipe straps which connect the pipe to the pipe support, in order to demonstrate Class I compliance which was not in accordance with the design and licensing basis. The Class I requirements were based on UFSAR, Section 5.10.1.2, and Specification No. C-173(Q). The UFSAR, Section 5.10.1.2, does not specify the use of inelastic capacity for the straps which are considered catalog items. The capacity is based on a specified load capacity which is based on the strap maintaining its structural integrity with no permanent or plastic deformation allowed when subjected to the design loading. Specification No. C-173(Q) delineated requirements consistent with UFSAR, Section 5.10.1.2. The inspectors determined the use of an inelastic acceptance limits for pipe support straps did not meet Class I requirements. The license entered the issue into their CAP as CR-PLP-2017-05246, “Spray Pipe Support,” dated November 14, 2017. The licensee performed an analysis to establish reasonable assurance of operability and the inspectors with support from the Office from the Nuclear Reactor Regulation.

Analysis: The inspectors determined the licensee’s failure to meet Class I requirements for the CS pipe supports HC44-R884, HC44-R884.1, HC44-R884.2, HC44-R884.3, HC44-R884.4, HC44-R884.5 and HC44-R884.9 was contrary to 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” and was a performance deficiency. The finding was determined to be more-than-minor because the finding was associated with the Barrier Integrity Cornerstone attribute of design control and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Specifically, failure to comply with Class I requirements did not ensure the Pipe Supports HC44-R884, HC44-R884.1, HC44-R884.2, HC44-R884.3, HC44-R884.4, HC44-R884.5 and HC44-R884.9 would function during a Class I design basis event and would adversely affect the CS piping system and containment barrier. The inspectors determined the finding could be evaluated using the Significance Determination Process (SDP) in accordance with Inspection Manual Chapter (IMC) 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” dated June 19, 2012, Exhibit 3, “Barrier Integrity Screening Questions,” for the Barrier Integrity cornerstone (Reactor Containment).

The inspector answered “no” to the Barrier Integrity questions for Reactor Containment. The finding screened as having very-low safety significance (Green).

The inspectors determined there was no cross-cutting aspect associated with this finding because the deficiency was a legacy design calculational issue and, therefore, was not indicative of licensee’s current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” requires, in part, that measures be established to ensure the applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The design control measures shall provide for verifying or checking the adequacy of design.

Contrary to the above, as of November 14, 2017, the design control measures failed to conform to Class I requirements and also failed to verify the adequacy of the design. Specifically, Calculation No. EA-SP-03369-02, Revision 0, failed to verify the adequacy of the design for the CS pipe supports HC44-R884, HC44-R884.1, HC44-R884.2, HC44-R884.3, HC44-R884.4, HC44-R884.5 and HC44-R884.9 to ensure it met the Class I requirements. Because this violation was of very-low safety significance (Green) and it was entered into the licensee’s CAP as CR-PLP-2017-05246, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000255/2017007-01; Containment Spray Pipe Support Strap Deficiencies)**

(2) Containment Dome Truss Analysis

Introduction: The inspectors identified an unresolved item (URI) concerning the analysis that demonstrated the design adequacy of the containment dome truss under design and licensing basis loading conditions.

Description: The dome truss system was originally designed to support the containment liner plate and wet concrete during the construction of the containment dome (i.e., the liner plate initially acted as a form and the truss supported the form). After the concrete cured, the dome truss system was lowered away from the liner and was used to support the safety injection tanks (SITs) and CS system piping and their associated supports. The CS and SIT systems are both safety-related which were required to be evaluated for seismic loads (self-weight and externally applied loads). The dome truss system would have also been required to be evaluated for seismic loads.

The UFSAR, Section 6.1, described the safety-related design function of the SIT system was to prevent fuel and cladding damage that could interfere with adequate emergency core cooling, and to limit the cladding-water reaction to less than approximately 1 percent for all break sizes in the primary system piping up to and including the double-ended rupture of the largest primary coolant pipe, for any break location, and for the applicable break time. Also, the SIT system also functions to provide rapid injection of large quantities of borated water for added shutdown capability during rapid cooldown of the primary system caused by a rupture of a main steam line.

The UFSAR Section 6.2.1 described the safety-related design function of the CS system was to limit the containment building pressure rise and reduce the airborne radioactivity in containment by providing a means for spraying the containment atmosphere after occurrence of a LOCA or a main steam line break.

The inspectors requested the design basis analysis of the dome truss system that considers the LOCA loading on the dome truss system as well as the seismic loading due to the applied design loads from the CS and SIT system. During the time of the inspection, the licensee was unable to locate the dome truss analysis.

In response to the inspectors concern, the licensee entered the issue into their CAP as CR 2017-05016, "Dome Trusses," dated November 1, 2017. The licensee is investigating the containment dome truss analysis further with the vendor of the dome truss system.

This issue is a URI pending additional inspector review of the design basis analysis for the containment dome truss system. **(URI 05000255/2017007-02; Containment Dome Truss Analysis)**

.6 Operating Procedure Accident Scenarios

a. Inspection Scope

The team performed a detailed reviewed of the procedures listed below. The procedures were compared to UFSAR, design assumptions, and training materials to assess their consistency. The following operating procedures were reviewed in detail:

- 4.48, "Time Critical Action/Time Sensitive Action Program Standard," Revision 6;
- EOP TCA, "EOP Time Critical/Time Sensitive Operator Action Basis," Revision 2;
- SOP 22, "Emergency Diesel Generators," Revision 74;
- AOP 41, "Alternate Safe Shutdown Procedure," Revision 3; and
- AOP Supplement 8, "Operation of Panels EC-150/EC-150A," Revision 0.

For the procedures listed, time dependent operator actions were reviewed for adequacy. This review included walk downs of in-plant actions with a licensed operator. In addition, the team evaluated operations interfaces with other departments such as engineering. The following operator actions were reviewed:

- Time critical operator actions to switch Control Room HVAC to Emergency Mode.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA2 Identification and Resolution of Problems

.1 Review of Items Entered Into the Corrective Action Program

a. Inspection Scope

The team reviewed a sample of problems identified by the licensee associated with the selected samples and that were entered into the CAP. The team reviewed these issues to verify an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions related to design issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problem into the CAP. The specific corrective action documents sampled and reviewed by the team are listed in the attachment to this report.

The team also selected seven issues identified during previous component design basis inspections to verify that the concerns were adequately evaluated and corrective actions were identified and implemented to resolve the concern, as necessary. The following issues were reviewed:

- NCV 05000255/2014008-02, “Undersized Supply Cables from Startup Transformer to 2400V Buses”;
- NCV 05000255/2014008-03, “Undersized Motors”;
- NCV 05000255/2014008-05, “Lack of Analysis for Electrical Containment Penetration Protection”;
- NCV 05000255/2014008-04, “Failure to Ensure that 480VAC System Voltages do not exceed Equipment Ratings”;
- NCV 05000255/2014008-09, “Failure to Include the Degraded Voltage Channel Time Delay in Technical Specification Surveillance Requirements”;
- NCV 05000255/2014008-10, “Failure to include the Degraded Voltage Time Delay in TS Surveillance Requirements”; and
- NCV 05000255/2014008-13, “Non-Conservative Surveillance for Emergency Diesel Generator Largest Load Reject Test.”

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Interim Meeting Summary

On November 16, 2017, the team presented the preliminary inspection results to Mr. J. Hardy and other members of the licensee staff. The licensee acknowledged the issues presented. The team confirmed that several documents reviewed were considered proprietary and were handled in accordance with the NRC policy related to proprietary information.

On November 17, 2017, the team presented the original inspection results to Mr. D. Corbin and other members of the licensee staff. The licensee acknowledged the issues presented. The team asked the licensee whether any materials examined during the inspection should be considered proprietary. Several documents reviewed by the team were considered proprietary information and were either returned to the licensee or handled in accordance with NRC policy on proprietary information.

.2 Exit Meeting Summary

On November 13, 2018, the revised inspection results were presented to Mr. C. Arnone. The licensee personnel acknowledged the inspection results presented and did not identify any proprietary content. The inspectors confirmed that all proprietary material reviewed during the inspection was returned to the licensee staff or handled in accordance with NRC policy on proprietary information.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

C. Arnone, Vice President
D. Corbon, General Manager Plant Operations
O. Gustafson, Regulatory Assurance and Performance Improvement Director
K. O'Connor, Engineering Director
J. Hardy, Regulatory Assurance Director
B. Sova, Design Engineering Manager
B. Baker, Maintenance Manager

U.S. Nuclear Regulatory Commission

M. Jeffers, Branch Chief
J. Benjamin, Senior Reactor Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000255/2017007-01	NCV	Containment Spray Pipe Support Strap Deficiencies (1R21.5.b(1))

Open

05000255/2017007-02	URI	Containment Dome Truss Analysis (1R21.5.b(2))

Discussed

None

LIST OF DOCUMENTS REVIEWED

The following is a list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspectors reviewed the documents in their entirety, but rather, that selected sections of portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document on this list does not imply NRC acceptance of the document or any part of it, unless this is stated in the body of the inspection report.

50.59 SCREENINGS

- 50.59 Screening 96-2716, Replace Pumps 18A/B with New Pumps; 12/15/1996

CALCULATIONS

- EA-ELEC-EDSA-06; Palisades AC Power System – Short Circuit Analysis; Revision 2
- EA-ELEC-EDSA-10; DC System Battery D01 EDSA Model Development and Load Flow Analysis; Revision 1
- EA-ELEC-EDSA-11; DC System Battery D02 EDSA Model Development and Load Flow Analysis; Revision 2
- EA-ELEC-VOLT-051; MCC Power Circuit Minimum Required Voltage Analysis; Revision 2
- EA-ELEC-LDTAB-009; Battery Sizing for the Palisades Class 1E Station Batteries D01 and D02; Revision6
- EA-SP-03374-01; Main Steam Piping Analysis Piping from Steam Generator E-50A to Containment Penetration 2; Revision 2
- EA-POC0007899-T2; Evaluation of Condensate Storage Tank T-2; Revision 1
- EA-SP-03370; Piping Analysis for Containment Spray Piping; Revision 2
- EA-SP-03369-01; Piping Evaluation for Containment Spray Piping; Revision 2
- EA-SP-03369-02; Piping Evaluation for Containment Spray Piping; Revision 0
- EA-SP-05904-01; Pipe Stress Analysis Auxiliary Feedwater Piping; Revision 4
- EA-SP-05901-01; Pipe Stress Analysis Auxiliary Feedwater Pump Suction; Revision 0
- EA-SP-03356-01; Auxiliary Feedwater Pump Suction; Revision 0
- EA-SP-03342-02; Piping Stress Analysis for 2” Auxiliary Feedwater Pump P-8A and B Recirculation Piping; Revision 0
- EA-SP-03342-01; Auxiliary Feedwater Discharge Piping; Revision 0
- EA-EC8083-01; Evaluation of CST for Tornado Loads; Revision 1
- EA-T-343-03; Determination of the Fuel Oil Transfer Pump Rates to the Diesel Generator Day Tanks; 05/12/1994
- EA-C-PAL-98-1748-02; Evaluation of Allowable Leakage Rate from the Spent Fuel Pool Cooling System; 03/15/1999
- EA-SC-96-051-01; Fuel Oil Transfer Pump Replacement; 03/06/1997
- EA-FC-958-05; Hydraulic Analysis for P-18A/B Pump Replacements; 03/06/1997
- EA-EC6432-01; Palisades Emergency Diesel Generator Diesel Fuel Oil Storage Requirements; 05/24/2010
- EA-E-PAL-94-010-01; Alternate Diesel Generator Air Driven Diaphragm Pump Flow; 06/09/1994
- EA-EC7120-01; Auxiliary Feedwater Pumps Low Suction Pressure Trips—Setpoint Change; 01/05/2009
- EA-A-PAL-94-095; Auxiliary Feedwater Pumps Net Positive Suction Head; 06/10/1994
- EA-ELEC-EDSA-03; LOCA with Offsite Power Available; Revision 2
- EA-ELEC-EDSA-04; Second Level UV Relay Setpoint Determination; Revision 0
- EA-ELEC-EDSA-06; Short Circuit Analysis; Revision 2

- EA-ELEC-LDTAB-005; EDG Steady State Loading Calculation; Revision 10
- EA-ELEC-VOLT-037; Degraded Voltage Calc for Safety-Related MOVs; Revision 3
- EA-ELEC-VOLT-050; MCC Control Circuit Voltage Analysis; Revision 3
- EA-ELEC-VOLT-051; MCC Power Circuit Required Voltage Analysis; Revision 2
- EA-ELEC-VOLT-052; DC Voltage Analysis; Revision 0
- 1D/202/151; Protection Calc—Bus 1D Incoming Breaker; Revision 1
- 1D/203/151; Protection Calc—Bus 1D Incoming Breaker; Revision 2
- 11-12/9B; Protection Calc—Load Center 12 Low Side; Revision 0
- 1D/201/150-151; Protection Calv—Station Power; Revision 4
- EA-GL8910-01; GL 89-10 MOV Thrust Window Calculations; Revision 11
- 1/9C; High Pressure Injection MOV MO-3009; Revision 2
- 1/4C; Low Pressure Injection MOV MO-3010; Revision 2
- EA-POC0007899; Roof Drain Pipe Analysis K6AB-4; Revision 6
- EA-ELEC-AMP-030; Capability of the 2400 V Feeder Calcs to Buses 1C and 1D from Startup Transformer 1-2; Revision 2

CORRECTIVE ACTION PROGRAM DOCUMENTS INITIATED DURING INSPECTION

- 2017-04610
- 2017-05110
- 2017-05124
- 2017-05016
- 2017-05246
- 2017-05014
- 2017-05256
- 2017-05237
- 2017-05015
- 2017-04995
- 2017-05240
- 2017-05247
- 2017-05232
- 2017-05288
- 2017-05264
- 2017-05264
- 2017-05265
- 2017-05282
- 2017-05283
- 2017-05265
- 2017-05264
- 2017-05251
- 2017-05240
- 2017-05232
- 2017-05076

CORRECTIVE ACCTIONS PROGRAM DOCUMENTS REVIEWED

- 2014-09030
- 2016-00798
- 2016-01740
- 2012-04164
- 2017-01248
- 2016-04972
- 2015-03116
- 2014-02381
- 2013-04050
- 2013-01381
- 2011-05337
- 2017-03793
- 2017-01588
- 2017-01422
- 2016-04008
- 2015-04160
- 2015-02056
- 2015-01737
- 2013-05039
- 2010-06100
- 2017-04248
- 2017-03007
- 2017-02667
- 2017-01642
- 2017-01249
- 2015-01841
- 2014-02899
- 2014-01437
- 2013-05294
- 2013-02802
- 2013-02764
- 2012-04613
- 2015-01803
- 2008-01616
- 2013-03392
- 2012-05719
- 2017-02655
- 2016-00026
- 2014-04902
- 2014-04679
- 2014-04680
- 2014-04696
- 2014-04864
- 2014-04450
- 2014-04860
- 2012-01245
- 2012-06773
- 2014-04903
- 2012-00004
- 2012-06773
- 2012-01245
- 2016-00026
- 2012-03818

DESIGN BASIS DOCUMENTS

- DBD-1.09; Design Basis Documents for the Main Steam System; Revision 4
- DBD-2.07; Design Basis Document for Spent Fuel Pool Cooling System ; Revision 5
- DBD-5.01; Diesel and Auxiliary System; Revision 7
- DBD-5.03; Design Basis Document for Emergency Diesel Generator Performance Criteria; Revision 9
- DBD-1.04; Design Basis Document for Chemical Volume Control System; Revision 7

DRAWINGS

- C-138; Containment Liner Support Trusses; Revision 9
- C-246; Reactor Building Safety Injection Tank Supports; Revision 4
- E-87, Sh.6; Schematic Diagram, CST Level and Alarm Indication; Revision 10
- E-100, Sh. 1; Schematic Diagram 480 V MCC Combination Starter and Feeders; Revision 28
- E-128, Sh.1; Schematic Diagram Charging Pump P55A Feeder Breaker Internals; Revision 5
- E-238, Sh. 1; Schematic Diagram Main Steam Isolation Valves; Revision 27
- E-257, Sh.1; Schematic Diagram Charging Pump P55A; Revision 24
- E-376, Sh.1; Conduit and Tray Plan for CST Instrument Line Freeze Protection; Revision 31
- E-679, Sh. 1; Schematic Diagram Diesel Oil Transfer Pumps; Revision 22
- E-897; Wiring Diagram, Freeze Protection Panel C100 & C100A; Revision 15
- M-202; P&ID Replacement Heat Tracing for CVC System; Revision 16
- M-205, Sh. 1; Connection Diagram SV-505A & B, Panel C-180 Piping & Instrument Diagram Main Steam & Auxiliary Turbine Systems; Revision 94
- M-214, Sh. 1; Piping & Instrument Diagram, Lube Oil, Fuel Oil & Diesel Generator Systems; Revision 81
- M-221 Sheet 2; Piping and Instrumentation Diagram Spent Fuel Pool Cooling System; Revision 61
- M-214; Piping and Instrument Diagram Lube Oil, Fuel Oil and Diesel Generator Systems; Revision 81
- M-221 SHT 2; Piping and Instrumentation Diagram Spent Fuel Pool Cooling System; Revoision 61
- C-92; Auxiliary Building Fuel Pool Liner Plate Details; Revision 8
- C-111; Auxiliary Building Spent Fuel Pool Sections and Detail; Revision 6
- C-110; Auxiliary Building Spent Fuel Pool Sections and Detail; Revision 7
- M-207; 0002; Piping and Instrument Diagram, Auxiliary Feedwater System; Revision 41
- M-214; Piping and Instrument Diagram, Lube Oil, Fuel Oil, and Diesel Generator Systems; Revision 41
- E-1 SHT A; Station Key Diagram; Revision 14
- E-4 SHT 1; Single Line Diagram 480 V Load Centers; Revision 45
- E-4 SHT 2; Single Line Diagram 480 V Load Centers; Revision 41
- E-5 SHT 1; Single Line Diagram 480 V Motor Control Centers; Revision 59

ENGINEERING CHANGES

- EC 58140; Install Permanent Shielding on Letdown Heat Exchanger E-58; Revision 0
- EC 58141; Install Shielding on Pressurizer Surge Line E-50A Platform; Revision 0
- EC 74734; Operability Input for Containment Dome Truss, Containment Spray and Safety Injection Relative to CR-2017-5016
- EC 74877; Input for Operability of Containment Spray Supports HC44-R884, HC44-884.1, HC44-R884.2, HC44-R884.3, HC44-R884.4, HC44-R884.5, and HC 44-R884.9; Revision 0

- EC 71766; Replacement of Diesel Fuel Oil Tank, T-10; Revision 0
- EA-SC-96-051-01; Fuel Oil Transfer Pump Replacement; Revision 1
- EC 56644; Diesel Generator Fuel Oil Tank to Comply with Michigan Fire Code; 05/18/2015
- EC 49797; Flex EC#22; Revision 0
- EC 47340; Flex EC#7; Revision 0
- EC 5000122470; Fast Bus Transfer Modification to Resolve TIA 2007-002; Revision 0

MISCELLANEOUS

- EDS Nuclear Report 02-0660-1087; Seismic Evaluation of Safety Injection Tank for Palisades Nuclear Plant; Revision 1
- ENN-DC-152; Preparation, Revision, Review, and Approval of Design Basis Documents; Revision 8
- Program SEP-AOV-PLP-001; Palisades Nuclear Power Plant Air Operated Valve Program Entergy Nuclear Engineering Programs; Revision 1
- Specification No. C-173(Q); Technical Requirements for the Analysis and Design of Safety-Related Pipe Supports; Revision 6
- M0120 0009; Laurence, RG Co. Inc. Information Bulletin for Safety Shut-Off 2-Way Manually Reset Solenoid Valves; Revision 0
- VTD-0010-0140; General Electric Instructions for Polyphase Induction Motors; Revision 0
- VTD-0660-0052; Vendor Manual, Rosemount Pressure Transmitter; Revision 0
- PLP-RPT-12-00026; Maintenance Rule Scoping Document—Spent Fuel Pool Cooling; Revision 1
- PLP-RPT-12-00026; Maintenance Rule Scoping Document—Auxiliary Building; Revision 1
- PLP-RPT-12-00026; Maintenance Rule Scoping Document—Chemical Volume Control System; Revision 1
- PLP-RPT-12-00026; Maintenance Rule Scoping Document—Fuel Oil System ; Revision 1
- PLLP-ESPO-PBSO-VAS; Palisades Basic HVAC System Orientation; Revision 4
- Spent Fuel Pool System Health Report; 10/05/2017
- Chemical Volume Control—Charging/Letdown System Health Report; 10/20/2017
- System Health Report—Switchyard System Q2-2017
- System Health Report—2400 VAC System Q3-2017
- System Health Report—480 VAC System
- L-HU-06-010; Response to GL 2006-002; 4/3/2006
- Response to RAI Regarding GL 2006-002; 1/29/2007
- SEP-MOV-PLP-001; MOV Program; Revision 2
- E0005-SH-0149-0000; Vendor Manual—Siemens Vacuum Circuit Breakers
- Docket 50-265—License DPR-20; Palisades Inservice Inspection Program Submittal of Relief Request No. 14; Revision 1
- Palisades Plant—Alternative to Defer Repair of Spent Fuel Pool Heat Exchanger E-53A Nozzle Weld; 04/14/2000
- A-PAL-98-072; Lack of Procedural Guidance for Placing SFP Cooling on Emergency Power; 09/01/1998
- G-ME-A39; Spent Fuel Pool Cooling System—Single Failure Analysis; 11/10/1976
- EN-LI-119; Apparent Cause Evaluation for Historical Spent Fuel Pool Leakage Step Rise; 11/14/2010
- EAR-99-0081; CVCS Declassification; 3/29/1999
- Letter; Palisades Plant—Resolution of Unresolved Safety Issue (USI) A-46, Verification of Seismic Qualification of Equipment in Operating Plants; 09/25/1998

OPERABILITY EVALUATIONS

- 2017-5016; Containment Dome Truss; 11/02/2017;
- 2017-5246; Containment Spray Pipe Support Straps; 11/14/2017
- EN-OP-104 Attachment 9.5; Condition Report Operability Evaluation 2017-02655 and 2017 -0266; 5/24/2017
- EN-OP-104 Attachment 9.5; Condition Report Operability Evaluation 2013-02855; 07/03/2013

PROCEDURES

- Procedure No. QO-37; Palisades Nuclear Plant Technical Specification Surveillance Procedure Main Steam Isolation and Bypass Valve Testing; Revision 13
- ENN-DC-152; Preparation, Revision, Review, and Approval of Design Basis Documents; Revision 8
- EM-04-58; Spent Fuel Pool METAMIC™ Coupon Surveillance Program; Revision 2
- AOP-30; Loss of Shutdown Cooling; Revision 0
- AOP-35; Loss of Service Water; Revision 0
- AOP-41; Alternate Safe Shutdown Procedure; Revision 3
- SOP-24; Ventilation and Air Conditioning System; Revision 75
- EN-DC-150; Condition Monitoring of Maintenance Rule Structures; Revision 13
- RO-28; Control Room Envelope Positive Pressure; Revision 31
- RT-2-2; Control Room HVAC Heat Removal Capability; Revision 14
- MO-33; Control Room Ventilation Emergency Operation; Revision 26
- SOP-27; Fuel Pool System; Revision 72
- AOP-26; Loss of Spent Fuel Pool Cooling; Revision 3
- WI-SFP;O-01; Spent Fuel Pool Cooling Pump Oil Sample; Revision 1
- SOP-2B; Chemical and Volume Control Purification and Chemical Injection; Revision 53
- SOP-2A; Chemical Volume Control System Standard Design Process; Revision 87
- EN-DC-115; Engineering Change Process; Revision 21
- EN-DC-105; Configuration Management; Revision 4
- MO-8A-1; Emergency Diesel Generator 1-1; Revision 96
- SOP-22; Emergency Diesel Generators; Revision 74
- AOP-38; Acts of Nature; Revision 11
- CVCO-4; Periodic Test Procedure—Charging Pumps; Revision 11
- RT-71H; Spent Fuel System Class 3 Inservice Test; Revision 9
- WI-SFP-O-01; Spent Fuel Pool Cooling Pump Oil Sample; Revision 1
- DWO-1; Operator Daily/Weekly Items Modes 1,2,3, and 4; Revision 108
- MC-17; Fuel Oil Sampling; Revision 33
- SEP-PLP-IST-102; Inservice Testing of Selected Safety Related Pumps; Revision 3
- EN-DC-153; Preventative Maintenance Component Classification; Revision 15
- EN-DC-310; Predictive Maintenance Program; Revision 8
- SEP-VIB-PLP-001; Palisades Vibration Monitoring Program; Revision 2
- RO-112; Reactor Head/Pressurizer Vent Flow Check; Revision 11
- EN-LI-118; Causal Evaluation Process; Revision 24
- SOP-23, Att 9; Cold Weather Checklist-Electrical; Revision 58
- RI-125; CST Level Instrument Calibration; Revision 13
- EOP-3.0; Station Blackout Recovery; Revision 18
- ARP-7; Auxiliary Systems Scheme EK-11; Revision 101
- SPS-E-20; Maintenance for 2400V Siemens Switchgear; Revision 7
- SPS-E-28; Safeguards Transformer 1-1 Load Tap Transformer 1-1; Revision 8
- SPS-E-27; Inspection and Testing of Safeguards Transformer 1-1; Revision 8

- SOP-32; 345KV Switchyard Operating Procedure; Revision 38

PROGRAMS

- SEP-AOV-PLP-001; Palisades Nuclear Power Plant Air Operated Valve Program Entergy Nuclear Engineering Programs; Revision 1

SURVEILLANCES/TESTING

- Fuel Oil Transfer Pump 18A IST Data; 2014–November 2017
- P-55A Vibration Data; 2014–November 2017
- RE-138; Calibration of Bus 1D Undervoltage and Time Delay Relays; Revision 15

WORK ORDERS/REQUEST

- WO 52655743; QO037 Main Steam Isolation and Bypass Valve Testing; 05/11/2017
- WO 52565880; QO037 Main Steam Isolation and Bypass Valve Testing; 10/10/2015
- WO 375934; QO037; Main Steam Isolation and Bypass Valve Testing; 06/21/2014
- WO 52751091; Emergency Diesel Fuel Oil Transfer Pump Test; 10/05/2017
- WO 00355289; P-18A, Troubleshoot and Correct Air In-Leakage; 08/23/2016
- WO 52574758; P-18A, Coupling Lubrication PM; 04/28/2016
- WO 52458340; RT-71H – Spent Fuel Pool System Class 3 Inservice Test; 09/28/2015
- WO 51628139; P-51B, Coupling Lubrication PM; 10/31/2017
- WO 5168138; P-51A, Coupling Lubrication PM; 10/31/2017
- WO 51627309; P-51B, Pump Bearing Oil Change; 10/31/2017
- WO 516273308; P-51A, Pump Bearing Oil Change; 10/31/2017
- WO 429141-01; RT-8C Engineered Safeguards EDG 1-1; 06/07/2017
- WO 327612-03; EDG 1-1 Voltage Regulator Replacement; 11/05/2007
- WO 52668400-01; Safeguards Transformer Load Tap Changer Maintenance; 09/10/2015
- WO 52578266; PM for Breaker 152-107; 02/04/2016;
- WO 52585445; DVR Time Surveillance 162-154; 03/21/2016

LIST OF ACRONYMS USED

AFW	Auxiliary Feedwater
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
CR	Condition Report
CS	Containment Spray
CST	Condensate Storage Tank
EC	Engineering Change
EDG	Emergency Diesel Generator
IEEE	Institute of Electrical and Electronic Engineers
IMC	Inspection Manual Chapter
LERF	Large Early Release Frequency
LOCA	Loss-of-Coolant Accident
LOOP	Loss of Off-Site Power
NCV	Non-Cited Violation
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
RG	Regulatory Guide
SIT	Safety Injection Tank
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item