UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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LISLE, IL 60532-4352

December 2, 2014

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

SUBJECT: PALISADES NUCLEAR PLANT COMPONENT DESIGN BASES INSPECTION 05000255/2014008

Dear Mr. Vitale:

On November 4, 2014, the U.S. Nuclear Regulatory Commission (NRC) completed a Component Design Bases Inspection (CDBI) inspection at your Palisades Nuclear Plant. The enclosed report documents the results of this inspection, which were discussed on November 4, 2014, with you, and other members of your staff.

Based on the results of this inspection, ten NRC-identified findings of very low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy. Additionally, one licensee-identified violation is listed in Section 4OA7 of this report.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Palisades Nuclear Plant. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Palisades Nuclear Plant.

In accordance with Title 10, *Code of Federal Regulations* (CFR) "Rules of Practice," a copy of this letter and its enclosure a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records System (PARS) component of NRC's Agencywide Documents Access and Management System (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Ann Marie Stone, Chief
Engineering Branch 2
Division of Reactor Safety

Docket No. 50-255
License No. DPR-20

Enclosure:
Inspection Report 05000255/2014008
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	05000255/2014008
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	September 8, 2014 through November 4, 2014
Inspectors:	S. Sheldon, Senior Engineering Inspector, Lead J. Corujo-Sandin, Engineering Inspector, Mechanical C. Brown, Engineering Inspector, Electrical C. Zoia, Operations Engineer L. Rodriguez, Engineering Inspector, Observer H. Leake, Electrical Contractor C. Baron, Mechanical Contractor
Approved by:	Ann Marie Stone, Chief Engineering Branch 2 Division of Reactor Safety

Enclosure

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SUMMARY OF FINDINGS

Inspection Report 05000255/2014008, 9/8/2014 – 11/4/2014, Palisades Nuclear Plant Component Design Bases Inspection (CDBI).

The inspection was a 3-week onsite baseline inspection that focused on the design of components. The inspection was conducted by regional engineering inspectors and two consultants. Ten Green findings were identified by the inspectors. The findings were considered Non-Cited Violations (NCVs) of NRC regulations. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using IMC 0609, "Significance Determination Process" dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas" effective date January 1, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process" Revision 5, dated February 2014.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding having very low safety significance and an associated Non-Cited Violation (NCV) of 10 CFR Part 50.36(c)(3), "Surveillance Requirements," for the failure to ensure the channel time delay for the degraded-voltage monitor was included in Technical Specification (TS) Surveillance Requirement (SR) 3.3.5.2.a. Specifically, the licensee failed to include in the TS SR the required time delay after the voltage relay trips before the preferred source of power is isolated and 1E electrical loads transferred to the stand-by Emergency Diesel Generators (EDGs). This finding was entered into the licensee's Corrective Action Program and the licensee's preliminary verification determined the degraded voltage monitors were still operable but degraded or non-conforming.

  The performance deficiency was determined to be more than minor because if left uncorrected, it would have the potential to lead to more significant safety concern. Specifically, by not incorporating the total time delay requirements into the Technical Specifications, (TS) the time could be changed without going through the TS change process, possibly leading to spurious trips of offsite power sources or possibly exceeding the accident analysis time is the FSAR. The inspectors determined the finding was of very low safety significance (Green) because it did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of the licensee's present performance. (Section 1R21.3.b(9))

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" for the failure to ensure the safety-related Engineered Safeguard Systems trains would not be adversely affected by air entrainment when aligned to the Safety Injection and Refueling Water (SIRW) Tank. Specifically, calculation EA-C-PAL-0877D, assumed incorrectly only one train of the Engineered Safeguards System (ESS) was in operation when evaluating if the SIRW Tank reaches the limit for critical submergence during a tank drawdown. As part of their corrective actions, the licensee re-evaluated the scenarios of concern, performed an operability evaluation, and implemented compensatory actions.

The performance deficiency was determined to be more than minor because it impacted the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, air entrainment into the ESS systems could potentially impact the operability of the system by air binding the pumps, reduce discharge flow, discharge pressure and/or delay injection. The inspectors determined the finding was of very low safety significance (Green) because the finding was a deficiency affecting the design or qualification of a mitigating structure system or component (SSC) but the SSC maintained its operability. The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of the licensee's present performance. (Section 1R21.3.b(1))

- <u>Green.</u> The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure the incoming feeder cables from startup transformer 1-2 to 2400 V safety-related Buses 1C and 1D were sized in accordance with their design basis, as described in Palisades FSAR Section 8.5.2. Specifically, the licensee failed to ensure the ampacity of the cables was at least as high as their maximum steady-state current. The licensee entered this finding into their Correction Action Program and verified the operability of the cables.

  The performance deficiency was determined to be more than minor, because it impacted the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, cables were undersized with respect to the loading that would automatically occur as the result of a design basis accident. The inspectors determined the finding was of very low safety significance (Green) because the SSC maintained its operability and functionality. This finding had a crosscutting aspect in the area of Human Performance, associated with the Design Margin component, because the licensee did not ensure that equipment is operated and maintained within design margins, and margins are carefully guarded and changed only through a systematic and rigorous process. [H.6] (Section 1R21.3.b(2))

- <u>Green.</u> The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure electric motors are sized in accordance with the design basis, as discussed in Palisades FSAR Section 6.2.3.1. Specifically, the horsepower ratings of certain motors are less than power demands of their driven equipment, and they were not analyzed to ensure overheating would not occur. The licensee entered this finding into their Correction Action Program with a recommended action to analyze the effect of the condition, and has verified the operability of the motors.

  This performance deficiency was determined to be more than minor, because it impacted the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, motors serving loads with power demands in excess of the motor horsepower ratings were not analyzed to ensure that motor damage would not occur. The inspectors determined the finding was of very low safety significance (Green) because the SSC maintained its operability and functionality. This finding had a crosscutting aspect in the area of Human Performance, associated with the Design Margin component, because the licensee failed to ensure that equipment is operated within design margins, and margins are carefully guarded and changed only through a systematic and rigorous process. [H.6] (Section 1R21.3.b(3))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to ensure that voltages on the 480V system do not exceed equipment ratings. Specifically, the licensee increased the output voltage of the supply transformers to the 480V safety-related buses by 2.5 percent, but failed to ensure the resulting voltages would not exceed equipment ratings when the system is powered from the station power transformer or emergency diesel generator. The licensee entered this finding into their Correction Action Program and verified the operability of the affected equipment.

  The performance deficiency was determined to be more than minor, because it impacted the Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to verify or check the voltage increase on the 480V system to ensure the maximum allowable voltage would not exceed equipment ratings. The inspectors determined the finding was of very low safety significance (Green) because the affected SSCs maintained their operability and functionality. The inspectors did not identify a cross-cutting aspect associated with this finding, because the finding was not representative of the licensee's present performance. (Section 1R21.3.b(4))

- Green. The inspectors identified a finding of very low safety significance and associated Non-Cited Violation of Technical Specifications 5.5.7, "Inservice Testing Program," for the failure to perform comprehensive pump testing of Containment Spray Pump P-54A in accordance with the code of record. Specifically, the licensee did not rerun a comprehensive pump test, as required by the code's ISTB-6300 "Systematic Error" section. As part of their corrective actions, the licensee entered the issue into the Corrective Action Program, and determined the component remained operable.

  The performance deficiency was determined to be more than minor because it impacted the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, failing to perform testing as required could result in the degradation of the equipment being undetected. The finding screened as having very low safety significance because the finding was a deficiency affecting the design or qualification of a mitigating structure system or component (SSC) but the SSC maintained its operability. The findings had a cross-cutting aspect in the area of Problem Identification and Resolution, Evaluation, because the licensee failed to thoroughly evaluate the issue to ensure that resolutions address causes and extents of conditions commensurate with their safety significance. [P.2] (Section 1R21.3.b(5))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to have adequate acceptance criteria in the emergency diesel generator surveillance procedures. Specifically, the licensee failed to ensure the surveillance test procedures for the emergency diesel generator largest load rejection test bounded the power demand of the largest load, as required by Technical Specification SR 3.8.1.5. The licensee entered this finding into their Correction Action Program and verified the operability of the emergency diesel generators.

  The performance deficiency was determined to be more than minor, because it impacted the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems to respond to initiating events to prevent undesirable consequences. Specifically, the surveillance procedure error could result in acceptance of test results that did not satisfy Technical Specification SR 3.8.1.5 for rejection of a load greater than or equal to the emergency diesel generator's single largest predicted post-accident load. The inspectors determined the finding was of very low safety significance (Green) because the SSC maintained its operability and functionality. This finding had a cross-cutting aspect in the area of Human Performance, associated with the Resources component, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety. [H.1] (Section 4OA2.1.b(1))

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to correctly translate design valve leakage limits into the applicable test procedure. Specifically, the acceptance criterion for emergency core cooling system (ECCS)/containment spray (CS) recirculation isolation valves CV-3027 and CV-3056 had not been correctly adjusted to account for the higher differential pressure associated with ECCS operation under post-accident conditions. The licensee entered this finding into their Corrective Action Program to correct the valve leakage limit.

  The performance deficiency was determined to be more than minor because it impacted the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, leakage approaching the procedural values would exceed analyzed dose calculations. The finding screened as of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding did not have an associated cross-cutting aspect because it was not representative of present performance. (Section 1R21.3.b(6))

- Green. The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," for the licensee's failure to identify non-safety-related sub-components improperly supplied with safety-related valves. Specifically, ECCS/CS recirculation isolation valves CV-3027 and CV-3056, which were installed in 2007, were supplied with non-safety-related sub-components. These components were identified as non-safety-related on the vendor drawings. In addition, the licensee later installed a section of non-safety-related tubing on valve CV-3027 based on the incorrect vendor drawing. The licensee entered this finding into their Corrective Action Program to correct the valve drawings and replace the non-safety-related parts.

The performance deficiency was determined to be more than minor because it impacted the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee failed to identify non-safety-related sub-components improperly supplied with safety-related valves which would form part of the containment barrier under post-accident conditions. The finding screened as of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding did not have an associated cross-cutting aspect because it was not representative of the licensee's present performance. (Section 1R21.3.b(7))

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to establish an adequate test program for the Shutdown Cooling (SDC) Heat Exchangers (HXs) to demonstrate they can perform as designed. Specifically, the licensee failed to take actions to ensure the SDC HXs' heat transfer capability met its design bases, as assumed in design bases calculations.

  The performance deficiency was determined to be more than minor because it impacted the Design Control attribute of the Barrier Integrity Cornerstone and adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee failed to verify the SDC HXs heat transfer capability met their design bases, as assumed in design bases calculations, to limit containment temperatures and pressures during an event. The finding screened as of very low safety significance (Green) because the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding had an associated cross-cutting aspect, Conservative Bias, in the Human Performance cross-cutting area. Specifically, on several occasions when the licensee identified the need to perform testing and/or inspection of the SDC HXs, the licensee did not take actions because they did not believe any regulatory requirements or technical issues existed that required the testing and/or inspections. [H.14] (Section 1R21.3.b(8))

B. Licensee-Identified Violations

Violations of very low safety or security significance or Severity Level IV that were identified by the licensee have been reviewed by the NRC. Corrective actions taken or planned by the licensee have been entered into the licensee's Corrective Action Program (CAP). These violations and CAP tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R21 Component Design Bases Inspection (71111.21)

.1 Introduction

The objective of the component design bases inspection is to verify the design bases have been correctly implemented for the selected risk significant components and the 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 Probabilistic Risk Assessment (PRA) 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 inspectors used information contained in the licensee's PRA and the Palisades Standardized Plant Analysis Risk Model to identify a scenario to use as the basis for component selection. The scenario selected was a Large Break Loss of Coolant Accident (LBLOCA) and subsequent transfer to recirculation. Based on this scenario, a number of risk significant components were selected for the inspection.

The inspectors 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 modification, 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, NRC resident inspector input of problem areas/equipment, and system health reports. 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 inspectors also identified procedures and modifications for review that were associated with the selected components. In addition, the inspectors selected operating experience issues associated with the selected components.

This inspection constituted 20 samples as defined in Inspection Procedure 71111.21-05.

.3 Component Design

a. Inspection Scope

The inspectors reviewed the Final Safety Analysis Report (FSAR), Technical Specifications (TS), design basis documents, drawings, calculations and other available design basis information, to determine the performance requirements of the selected components. The inspectors used applicable industry standards, such as the American Society of Mechanical Engineers (ASME) Code, Institute of Electrical and Electronics Engineers (IEEE) 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 (GLs), Regulatory Issue Summaries (RISs), and Information Notices (INs). The review was to verify 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 include 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, and licensee Corrective Action Program documents. Field walkdowns were conducted for all accessible components to assess material condition and to verify that the as-built condition was consistent with the design. Other attributes reviewed are included as part of the scope for each individual component.

The following 14 components were reviewed:

- Containment Spray Pump P-54A: The inspectors reviewed calculations related to pump hydraulics to ensure the pump was capable of performing its design bases accident mitigation functions. The calculations included analysis of net positive suction head (NPSH), discharge head, flow, and management of potential air entrainment mechanisms. The inspectors also reviewed procedures and system diagrams to ensure the pump would be operated within its design parameters during an event. Recent Inservice Test (IST) reports and historical trends data were reviewed to assess compliance with Technical Specification surveillance requirements, compliance with the applicable code of record, review potential component degradation, and impact on design margins. The inspectors discussed the component's performance with the system engineer and reviewed recent corrective action documents related to the SSC in order to assess the overall health of the component. In addition, the inspectors performed a walkdown to assess material condition of the pump and supporting components.

- 2400V Switchgear 1D: In addition to the generic list of attributes listed above, the inspectors reviewed electrical diagrams, calculations, and procedures, including system short circuit and load flow calculations. Incoming breaker protective relay trip setpoints were reviewed to evaluate the adequacy of the switchgear bus and breakers to carry anticipated loads under limiting conditions and to withstand and interrupt maximum available faults. The inspectors also reviewed the voltage profile of the offsite system, voltage drop calculations, and the undervoltage relay settings to assess adequacy of voltage at the terminals of the safety-related loads and ability to remain connected to offsite power under worst operating and accident conditions. Sizing of the incoming feeder cables was reviewed to determine their capability under worst accident conditions.

- SIRW Tank Suction Valve (CV-3057): The inspectors reviewed procedures and system diagrams to ensure the component would be operated within its design parameters during a design basis event. The inspectors discussed the component's performance with the system engineer, and reviewed recent corrective action documents related to the SSC in order to assess the overall health of the component. Calculations related to the component's safety-related air supply was also reviewed. The inspectors reviewed the circuit protection and the environmental qualification to confirm the circuit was adequately protected, and the valve was capable of performing its intended safety function. Voltage drop calculations were reviewed to verify the associated control circuits had adequate voltage under degraded voltage conditions. The inspectors also reviewed surveillance test data, any operator control limitations, operating procedures, vendor and generic communications, recent CRs and operability evaluations for any anomalous indications or possible difficulties in system operation. In addition, the inspectors performed a walkdown to assess material condition of the valve, supporting components, and identify any potential concerns related to adverse interaction with the surrounding equipment.

- Containment Sump Outlet Valve CV-3029: The inspectors reviewed the design basis of the air-operated valve including thrust calculations, the basis for valve stroke time requirements, and the associated control logic. The inspectors reviewed test, and emergency procedures, as well as the response of the system to the failure of the valve to operate under accident conditions. The inspectors reviewed the capacity of the high-pressure air system to verify its capability to operate the valve with a loss of normal instrument air. The inspectors reviewed air system leak test procedures and results and performed a walkdown of the valve to verify its material condition. The inspectors also reviewed the circuit protection, the environmental qualification of affected circuit components to confirm the circuit was adequately protected, and the valve was capable of performing its intended safety function during a design basis accident. Voltage drop calculations were reviewed to verify the associated control circuits had adequate voltage under degraded voltage conditions. The inspectors also reviewed operating procedures, vendor and generic communications, recent CRs and operability evaluations for any anomalous indications or possible difficulties in system operation.

- SIRW Tank Level Switch LS-327: The inspectors reviewed installation drawings and vendor documentation to verify the switch was appropriate for its use. The inspectors reviewed the Recirculation Actuation Signal (RAS) logic to verify the switch met its design basis. The inspectors also reviewed switch calibration and testing procedures. In addition, the inspectors performed a walkdown of the SIRW tank to assess the level transmitters used for calibration of the switch.

- Shutdown Cooling Heat Exchanger E-60A: The inspectors reviewed heat exchanger design documents to verify assumptions made in design bases calculations. The inspectors reviewed normal, test, and emergency procedures. The inspectors also reviewed historical information to understand how heat exchanger performance had been validated since original installation. A walkdown was performed to assess the material condition of the heat exchanger and supporting components. In addition, the inspectors reviewed aging management aspects related to the heat exchanger.

- Component Cooling Water (CCW) Pump (P-52B): The inspectors reviewed calculations related to pump hydraulics to ensure the pump was capable of performing its design bases accident mitigation functions. Included in the review were cooling requirements for the component and the supplied loads. Also, reviewed were potential alternate sources for cooling the supplied load if CCW were to become unavailable during an accident. The inspectors also reviewed procedures and system diagrams to ensure the pump would be operated within its design parameters during an event. The inspectors discussed the component's performance with the system engineer, and reviewed recent corrective action documents related to the SSC in order to assess the overall health of the component. In addition, the inspectors performed a walkdown to assess material condition of the pump, supporting components, and identify any potential concerns related to adverse interaction with the surrounding or nearby equipment.

- Component Cooling Water (CCW) Relief Valve RV-0956: The inspectors also reviewed procedures and system diagrams to ensure the pump would be operated within its design parameters during an event. The inspectors discussed the component's performance with the system engineer, and reviewed recent corrective action documents related to the SSC in order to assess the overall health of the component. In addition, the inspectors reviewed vendor manuals and technical sheets for the component. The required licensing and design bases for the component were reviewed and discussed with NRR, and requires further evaluation.

- Containment Sump Strainer: The inspectors reviewed results of sump inspections and ECCS pump NPSH analyses to verify the pressure drop across the screens under post-accident conditions. The inspectors reviewed normal, test, and emergency procedures. The inspectors also reviewed the basis for the maximum particle size.

- 125 VDC Bus D10: The inspectors reviewed the circuit diagrams, the short circuit current calculation, and the coordination calculation to confirm the short circuit duty and the proper coordination between the panel fuses and branch circuit cabling with the upstream protective device. The inspectors also reviewed the panel electrical loading and voltage drop calculations, and the branch circuit cabling, to confirm bus and circuit cable ampacity was adequate and branch circuits had adequate voltage. The inspectors also reviewed recent CRs, operability evaluations, and operating procedures for any anomalous indications.

- LPSI Pump Suction Crosstie Valve MO-3090: The inspectors reviewed the design basis of the motor-operated valve including thrust calculations, the basis for valve stroke time requirements, and the associated control logic. The inspectors reviewed normal, test, and emergency procedures including the response to the single failure of an electrical power supply to MO-3090. The inspectors also reviewed the interface between the electrical system distribution calculations and the valve thrust calculation to verify adequate voltage to the valve motor under the most limiting conditions. The inspectors performed a walkdown of the valve to verify its material condition.

- 480V Motor Control Center (MCC) 23: In addition to the generic list of attributes listed above, the inspectors reviewed electrical diagrams, calculations, and procedures, including system short circuit calculations and main alternating current (AC) electrical distribution analysis. Voltage calculations were reviewed to verify minimum and maximum voltages were within equipment capabilities, including downstream power and control components. Thermal overload protection for motor operated valves was reviewed to assess the adequacy of valve motor overload protection.

- ECCS/CS Minimum Flow Valve CV-3027: The inspectors reviewed the design basis of the air-operated valve including thrust calculations, the basis for valve stroke time requirements, and the associated control logic. The inspectors reviewed normal, test, and emergency procedures as well as the response of the system to the single failure of the valve to operate under accident conditions. The inspectors reviewed the basis for the valve leakage limits and reviewed leak test procedures and results. The inspectors also reviewed the safety classification of valve sub-components. The inspectors performed a walkdown of the valve to verify its material condition.

- 125 VDC Bus D11: The inspectors reviewed the circuit diagrams, the short circuit current calculation, and the coordination calculation to confirm the short circuit duty and the proper coordination between the panel fuses and branch circuit cabling with the upstream protective device. The inspectors reviewed the degraded voltage protection design scheme to determine whether it afforded adequate voltage to safety-related devices at all voltage distribution levels. The inspectors also reviewed the panel electrical loading and the branch circuit cabling, to confirm bus, circuit cable ampacity was adequate, and branch circuits had adequate voltage. The inspectors also reviewed recent CRs, operability evaluations, and operating procedures for any anomalous indications.

b. Findings

(1) Failure to Ensure Engineered Safeguards Systems (ESS) Are Not Adversely Affected By Air Entrainment

Introduction: A finding of very low safety significance (Green) and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control" was identified by the inspectors for the failure to ensure the safety-related ESS Trains would not be adversely affected by air entrainment when the ESS trains are aligned to the SIRW tank under design bases accidents.

Description: During a postulated design bases accident, the licensee's ESS would be required to operate. At the beginning of the accident the two trains of ESS would begin to operate with their associated pumps' suction lined up to the SIRW Tank. Train A includes: one Low Pressure Safety Injection (LPSI) pump, one Containment Spray (CS) pump and one High Pressure Safety Injection (HPSI) pump. Train B includes one LPSI pump, two CS pumps and one HPSI pump. When the water level in the SIRW tank reaches a pre-determined low level, a Recirculation Actuation Signal (RAS) is generated. Once the RAS occurs, the LPCI pumps are tripped off and the pumps' suctions begin to automatically re-aligned to the containment sump. Per the licensee's design bases this process can take up to 70 seconds to occur. If the water level were to get too low on the SIRW tank before the automatic swap to the containment sump concludes, air could be drawn into the system. If enough air entrainment occurred, the operability of the downstream pumps could be affected. Air entrainment can degrade pump performance, causing the pumps to air bind, reduce discharge pressure/flow and/or cause delays in injection.

The inspectors reviewed design bases calculation EA-C-PAL-0877D, "Evaluation of the Potential for Excessive Air Entrainment Caused by Vortexing in the SIRW tank During a LOCA," Revision 1. In the calculation the licensee established the critical submergences needed before air entrainment mechanisms could develop (vortexing and radial inflow) and evaluated the lowest water levels reached in the SIRW tank during a design bases accident. The calculation determined the critical submergence limit for vortexing would be exceeded. The licensee estimated the maximum credible air entrainment at the pumps' suctions as 3.6 percent by volume. This value was determined using the "enveloping line" correlation based on Knauss, Jost, "Swirling Flow Problems at Intakes," A. A. Balkema, Rotterdam, Netherlands, 1987 and the expected void compression using the Ideal Gas Law. However, the inspectors noted the configuration described in the referenced material did not match the licensee’s SIRW tank outlet piping configuration. The referenced material was based on horizontal outlet pipes, not vertical as in the case of the licensee’s SIRW tank. Also, the expected void fraction at the pumps exceeded those considered acceptable per NEI 09-10, "Guidelines for Effective Prevention and Management of System Gas Accumulation.” The NEI 09-10 provides the void fraction acceptance criteria used for operability calls in Palisades' Gas Accumulation Management Program Document. This is discussed in Attachment 9.5 of EN-DC-219, "Gas Accumulation Management," Revision 3 (the licensee’s Gas Management Program Document):

> "These guidelines do not provide acceptance criteria to support permanent design bases or procedural changes. These guidelines do provide a toolset for establishing a Reasonable Expectation of Operability based on industry standards for non-conforming conditions associated with gas intrusion. Other tools may be utilized by a specific site if properly supported by site evaluation."

Upon further review, the inspectors noticed that when determining the lowest level reached in the SIRW tank, calculation EA-C-PAL-0877D did not assume both trains of ESS could be in operation and taking suction from the SIRW tank concurrently. In addition, the calculation did not identify that a postulated single failure could result in a failure of RAS to occur for one train. Under this condition, one train of ESS would not realign to the containment sump and the LPCI pump on the affected train would continue to operate. Both of these concerns would result in a SIRW tank level lower than that assumed in the calculation. The licensee documented the inspectors' concerns under CR-PLP-2014-04472 and CR-PLP-2014-04665. An operability evaluation was also developed under CR-PLP-2014-04665. In order to maintain operability, the licensee re-evaluated the scenarios in Question (EC 53093) and adjusted their acceptance criteria for critical submergence. In addition, compensatory actions to ensure the system behaved as expected in their re-valuation were needed. These actions included: (1) revising the acceptance criteria in QO-2 for the stroke times of the SIRW tank outlet isolation valves CV-3031 and CV-3057 (shortened the allowed stroke time), and (2) revising the acceptance criteria in RI-14 for the SIRW tank low level switches LS-0327, LS-0328, LS-0329 and LS-0330.

Analysis: The inspectors determined the licensee's failure to ensure the safety-related ESS Trains would not be adversely affected by air entrainment when aligned to the SIRW tank was a performance deficiency. Specifically, when evaluating if the SIRW tank reaches the limit for critical submergence (during a tank drawdown) calculation EA-C-PAL-0877D, assumed only one train of ESS was in operation instead of both trains. Additionally, the licensee did not evaluate the potential effects on the ESS system if both trains are assumed to be in operation, or if one train fails to receive its RAS signal. Both scenarios result in lower water levels in the SIRW tank, and potential air ingestion into the ESS trains.

The performance deficiency was determined to be more than minor because it impacted the Equipment Performance attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, air entrainment into the ESS systems could potentially impact the operability of the system by air binding the pumps, reduce discharge flow, discharge pressure and/or delay injection. The licensee performed an operability evaluation, and re-analyzed the SIRW tank drain down (EC 53093) in order to ensure the ESS systems remained Operable.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04," "Initial Characterization of Findings." Specifically, the inspectors used IMC 0609 Appendix A "SDP for Findings At-Power" Exhibit 2, "Mitigating Systems Screening Questions". The finding was a deficiency affecting the design or qualification of a mitigating structure system or component (SSC) but the SSC maintained its operability. Specifically, the licensee performed an operability evaluation CR-PLP-2014-04665 and determined the performance deficiency would result in lower SIRW Tank levels but at least one train of ESS would remain operable during a design bases accident. As a result, the finding screened as having very low safety significance, i.e., Green.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of the licensee's present performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of September 8, 2014, the licensee failed to ensure their ESS trains would not be adversely affected by air entrainment when these are aligned to the SIRW tank. Specifically, calculation EA-C-PAL-0877D, "Evaluation of the Potential for Excessive Air Entrainment Caused by Vortexing in the SIRW tank During a LOCA," Revision 1, failed to evaluate the potential effects on the ESS systems if both trains were assumed to be in operation, or if one train failed to receive its RAS signal. As part of their corrective actions, the licensee initiated CR-PLP-2014-04665, performed an operability evaluation, and implemented compensatory measures needed to ensure the Engineered Safeguard Systems are able to perform their required safety functions during design bases conditions.

Because this violation was of very low safety significance, and it was entered into the licensee's Corrective Action Program as CR-PLP-2014-04472 and CR-PLP-2014-04665, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-01, Failure to Ensure Engineered Safeguards Systems Are Not Adversely Affected By Air Entrainment.]

(2) Undersized Supply Cables from Startup Transformer to 2400V Buses

Introduction: A finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the licensee's failure to ensure the incoming feeder cables from startup transformer 1-2 to 2400 V safety-related Buses 1C and 1D were sized in accordance with their design basis, as described in Palisades FSAR Section 8.5.2. Specifically, the licensee failed to ensure the ampacity of the cables was at least as high as their maximum predicted steady-state current.

Description: On October 23, 2013, the licensee issued calculation EA-ELEC-EDSA-03, "LOCA with Offsite Power Available," Revision 2, which identified that the maximum steady-state load on Bus 1C during a LOCA would be 873 amps (Tables B2-1C and SU-CD-1C-B2) and on Bus 1D would be 953 amps (Table SU-1D-2355-B2). Palisades FSAR Section 8.5.2 states that "Cables installed in...underground ducts are thermally sized in accordance with NEC or IPCEA/ICEA ampacity values (depending on cable physical size) of concentric stranded insulated cable for the conductor operating temperature of the insulation." According to Section 4.0 a of Engineering Analysis EA-ELEC-AMP-030, "Capability of the 2400V Feeder Cables to Buses 1C and 1D from the Startup Transformer 1-2," Revision 2, the rated operating temperature of the cable insulation is 90°C. According to Engineering Change EC 24546 to Design Basis Document DBD 3.04, "2400V AC System," Revision 7, the ampacity is 636 amps (current that would raise the cable temperature to 90°C). Thus, the cable ampacity is exceeded by the loading levels calculated in EA-ELEC-EDSA-03, contrary to the requirement of FSAR Section 8.5.2. Although the licensee previously entered this issue into their Margin Management Program as Margin Issue 414, they failed to characterize it as a design basis deficiency.

Based on inspector concerns, the licensee entered this issue into the Corrective Action Program as CR- CR-PLP-2014-4860 with a recommended action to provide additional procedural guidance for unloading the cables if an overload condition were to occur. The licensee also verified current operability by confirming the cables would not fail as a result of the condition.

Analysis: The inspectors determined the failure to ensure the supply cables to Buses 1C and 1D are sized in accordance with the design basis 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 Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, cables were undersized with respect to the loading that would automatically occur as the result of a design basis event.

The inspectors assessed this finding for significance in accordance with NRC Manual Chapter 0609, Appendix A, Exhibit 2, The Significance Determination Process (SDP) for Findings At-Power, and determined that it was of very low safety significance (Green), because the SSC maintained its operability and functionality. This finding had a crosscutting aspect in the area of Human Performance, associated with the Design Margin component, because the licensee did not ensure that equipment is operated and maintained within design margins, and margins are carefully guarded and changed only through a systematic and rigorous process. The licensee's margin management program had not corrected this known condition. [H.6]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, "Measures shall be established to assure that 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, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program."

Contrary to the above, as of September 8, 2014, the licensee failed to assure that the design basis for cable sizing specified in FSAR Section 8.5.2 was translated into the design of the cables. Specifically, the design ampacity of the cables was less than their maximum predicted steady-state load. Because this violation was of very low safety significance, and it was entered into the licensee's Corrective Action Program as CR-CR-PLP-2014-4860 with a recommended action to provide additional procedural guidance for unloading the cables if an overload condition were to occur, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. The licensee also verified current operability by confirming the cables would not fail as a result of the condition. [NCV 05000255/2014008-02, Undersized Supply Cables from Startup Transformer to 2400V Buses.]

(3) Undersized Motors

Introduction: A finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the licensee's failure to ensure that electric motors are sized in accordance with the design basis, as discussed in Palisades FSAR Section 6.2.3.1. Specifically, the horsepower ratings of certain motors are less than power demands of their driven equipment, and they were not analyzed to ensure that overheating would not occur.

Description: Palisades FSAR Section 6.2.3.1 states the containment spray "motor drivers have been selected to be non-overloading over the entire pump operating range." To the contrary, on August 6, 2014, the licensee issued calculation EA-ELEC-LDTAB-005, "Emergency Diesel Generators 1-1 and 1-2 Steady State Loadings," Revision 10, which identifies, in Table C.1, that the motor for containment spray pump P-54A is rated 250 HP, but the power demand of the pump is 278 BHP. The calculation also lists other motors having ratings less than the power demand of their associated driven equipment. Power demands of the containment spray pumps and other motor driven equipment would be even higher than those listed in Calculation Tables A.1 through E.2 when the emergency diesel generator is operating at its maximum allowable frequency and voltage. The calculation identifies, in Section 4.24, that this operating condition would cause a 6.32 percent increase in the power demands. The inspectors noted calculation EA-ELEC-LDTAB-005 does not analyze or justify the undersized motor condition, nor is there any other analysis that justifies this condition. In the, "Quality Assurance Program Manual," Revision 26, the licensee is committed to Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants," Revision 2, which cites ANSI N45.2.11-1974, “Quality Assurance Requirements for the Design of Nuclear Power Plants.” ANSI N45.2.11 states, “Measures shall be applied to verify the adequacy of design. Design verification is the process of reviewing, confirming, or substantiating the design by one or more methods to provide assurance that the design meets the specified design inputs…. The results of design verification efforts shall be clearly documented.” Contrary to this requirement, the licensee had not verified and documented the thermal capability of motors serving loads in excess of the motor horsepower rating.

The licensee entered this issue into their Corrective Action Program as CR-PLP-2014-4902 with a recommended action to analyze the effect of the condition. The licensee subsequently verified operability of the motors.

Analysis: The inspectors determined the failure to ensure that electric motors are sized in accordance with the design basis and verified to be within their thermal capability 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 Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, certain motors serve loads with demands in excess of the motor horsepower ratings, and they were not analyzed to ensure that overheating would not occur.

The inspectors assessed this finding for significance in accordance with NRC Manual Chapter 0609, Appendix A, Exhibit 2, The Significance Determination Process (SDP) for Findings At-Power, and determined that it was of very low safety significance (Green), because the SSC maintained its operability and functionality.

This finding had a crosscutting aspect in the area of Human Performance, associated with the Design Margin component, because the licensee did not ensure that equipment is operated and maintained within design margins, and margins are carefully guarded and changed only through a systematic and rigorous process. [H.6]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, “Design Control” requires, in part, that “Measures shall be established to assure that 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, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.”

Contrary to the above, as of September 8, 2014, the licensee failed to verify the adequacy of design of certain motors. Specifically, the horsepower ratings of those motors did not bound the power demands of their driven equipment, and they were not analyzed to ensure that the condition would not cause overheating.

Because this violation was of very low safety significance, and it was entered into the licensee’s Corrective Action Program as CR-PLP-2014-4902, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-03, Undersized Motors.]

(4) Failure to Ensure that 480V System Voltages do not Exceed Equipment Ratings

Introduction: A finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," was identified by the inspectors for the licensee's failure to ensure that voltages on the 480V system do not exceed equipment ratings. Specifically, the licensee increased the output voltage of the supply transformers to the 480V safety-related buses by 2.5 percent, but failed to ensure that the resulting voltages would not exceed equipment ratings when the system is powered from the station power transformer or emergency diesel generator.

Description: On March 30, 1995, the licensee issued calculation EA-ELEC-VOLT-036, "Station Power Transformers No. 11, 12, 19, and 20 Tap Change - 2.5 percent," Revision 0, which states, in Section IV.2.1, "the maximum allowed voltage at 480 V Load Centers/MCC's is 1.058 pu (480 V base)." On July 28, 1995, the licensee issued Specification Change document SC-94-102, Revision 0, to increase the output voltage of the supply transformers to the 480V safety-related buses by 2.5 percent. The inspectors determined the licensee failed to recognize that this could result in voltages at the 480V buses in excess of the 1.058 per unit limit. In addition, calculation EA-ELEC-VOLT-036, Section IV.4.3, postulated a non-conservative maximum voltage on the 2400V buses of 1.0 per unit (2400V). However, voltages on the 2400V buses in excess of 1.0 per unit could occur due to 1) operation on the diesel generator supply within the diesel generator voltage range up to 1.05 per unit, as allowed by Palisades Technical Specification SR 3.8.1.2, or 2) operation on the non-regulated Station Power Transformer 1-2 supply during shutdown conditions, as discussed in Palisades Technical Specification Basis 3.8.2.

The inspectors were concerned these higher voltages on the 2400V buses, in conjunction with the 2.5 percent increase in voltage to the 480V buses, could result in over voltages on the 480V buses. Voltages higher than the 1.058 per unit calculation limit could cause exposure of load equipment to voltages in excess of the manufacturers' ratings and diminished capability of electrical protective devices to interrupt short circuit currents.

The licensee entered this issue into their Corrective Action Program as CR-PLP-2014-4696 with a recommended action to further analyze emergency diesel generator operation at its maximum allowable voltage and CR-PLP-2014-4864 with an action to restrict switchyard maximum voltage when the safety-related buses are powered from the station power transformer. The licensee subsequently determined that all equipment was operable at the full range of diesel operation.

Analysis: The inspectors determined the licensee's failure to verify or check the adequacy of Specification Change document SC-94-102, Revision 0, 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 Design Control attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to verify or check the adequacy of Engineering Assessment EA-ELEC-VOLT-036, "Station Power Transformers Nos. 11, 12, 19, and 20 Tap Change - 2.5 percent," Revision 0, to ensure that postulated maximum source voltages bounded allowable operating conditions and thus that the voltage increase would not result in overvoltage conditions.

The inspectors assessed this finding for significance in accordance with NRC Manual Chapter 0609, Appendix A, Exhibit 2. The Significance Determination Process (SDP) for Findings At-Power, and determined that it was of very low safety significance (Green), because the SSCs maintained their operability and functionality. The inspectors did not identify a cross-cutting aspect associated with this finding, because the finding was not representative of the licensee's present performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control" requires, in part, that "Measures shall be established to assure that 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, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program." Contrary to the above, as of September 8, 2014, the licensee failed to verify the adequacy of the design of the transformer tap settings. Specifically, the 2.5 percent boost in voltages to the 480V buses could result in voltages in excess of equipment ratings under certain allowable operating conditions.

Because this violation was of very low safety significance, and it was entered into the licensee's Corrective Action Program as CR-PLP-2014-4696, and CR-PLP-2014-4864 this violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. [NCV 05000255/2014008-04, Failure to Ensure that 480V System Voltages do not Exceed Equipment Ratings.]

(5) Failure To Perform Comprehensive Pump Testing Of Containment Spray Pump P-54A In Accordance With The Inservice Testing Program.

Introduction: A finding of very low safety significance (Green) and associated Non-Cited Violation of Technical Specifications 5.5.7, "Inservice Testing Program" was identified by the inspectors for the licensee's failure to perform comprehensive pump testing of Containment Spray pump P-54A in accordance with the code of record.

Description: On February 25, 2014 the licensee performed WO 52435851 01, "RO-98 - LPSI PMP VIB, CONT SPRY PMP VIB, DIS CK," which includes the comprehensive pump test of Containment Spray (CS) Pump P-54A. The comprehensive pump test is performed every 18 months in accordance with the ASME OMb Code-2001, through ASME OMb Code-2003 Addenda, which according to FSAR Section 6.9.2.1, "Pump Testing Program," and Site Engineering Program SEP-PLP-IST-102, "Inservice Testing of Selected Safety-Related Pumps" is the code of record for Palisades. Comprehensive testing of pumps is performed in order to detect potential degradation of the equipment and helps ensure the SSC will be able to meet its design requirements. During the test, the recorded differential pressure (DP) was 142.0 psid as read on differential pressure indicator DPI-0319A (range 0 to 250 psid). This value was within the Alert Range established by the licensee's procedure. Procedure RO-98 establishes the different criteria for P-54A as follows:

- The Acceptable Range was a 143.2 to 158.6 psid;
- The Alert Range was 138.6 to 143.2 psid;
- The Required Action Range was DP lower than 138.6 psid or greater than 158.6 psid; and
- The Acceptance Criteria was DP greater than 136.9 psid at 2250 +/- 35 gpm

The licensee documented the test results in the Alert Range under CR-PLP-2014-01679. The licensee suspected instrumentation issues may have been the cause of the deviation. Site personnel performed a historical search of RO-98 testing and identified several CRs related to instrumentation issues including CR-PLP-2010-05181 and CR-PLP-2010-06403. On January 13, 2014, the DPI-0319A had been calibrated to ensure it was within the comprehensive test required accuracy. On February 26, 2014, a post-test calibration was performed in accordance with WO 52435851 01. The result of the calibrations showed pressure indicator DPI-0319A was outside of the test requirements. The results indicated the gauge was reading lower than actual DP. The licensee determined the deviation was approximately 8.75 psid lower and concluded this would indicate an actual DP of approximately 150 psid under RO-98 conditions for pump P-54A. The licensee concluded this would result in a DP within the acceptable range. In addition to the above, the license performed an assessment of the vibration results obtained and evaluated various pump result trends (comprehensive and quarterly testing) associated with pump performance. Based on this review, documented on CR-PLP-2014-01679, the licensee concluded pump P-54A was actually operating within its acceptable range and no additional corrective actions were required.

However, the inspectors were concerned the comprehensive pump test performed on February 25, 2014, was invalid because it did not meet the requirements specified by the ASME code. Specifically, the requirements for differential pressure instrument accuracy. Per ASME OMb Code-2003, ISTB-3510(a) “Accuracy” states: “Instrument accuracy shall be within the limits of Table ISTB-3500-1.” Table ISTB-3500-1, “Required Instrument Accuracy,” requires the differential pressure instrument accuracy for differential pressure readings during a comprehensive test to be plus or minus ½ percent full scale. The post-test calibration results showed DPI-0319A did not meet the required accuracy. The ASME code, Section ISTB-6300 "Systematic Error" requires:

> "When a test shows measured parameter values that fall outside of the acceptable range of Table ISTB-5100-1, Table ISTB-5200-1, Table ISTB-5300-1, or Table ISTB-5300-2, as applicable, that have resulted from an identified systematic error, such as improper system lineup or inaccurate instrumentation, the test shall be rerun after correcting the error."

Containment Spray Pump, P-54A, acceptance values for testing fall under Table ISTB-5100-1 "Centrifugal Pump Test Acceptance Criteria.” As such, the licensee should have rerun the comprehensive test once DPI-0319A was shown to be out of calibration.

Failing to perform the IST comprehensive test as prescribe by the code was contrary to Technical Specifications (TS) 5.5.7, which requires an Inservice Testing Program be establish, implemented, and maintained. Section TS 5.5.7 a. establishes the required IST intervals as required by the code. Complete details for pumps IST are contained in Site Engineering Program SEP-PLP-IST-102, "Inservice Testing of Selected Safety-Related Pumps." The licensee documented the inspectors' concerns and documented the basis of operability under CR-PLP-2014-04881. The licensee determined the component remained operable based on the evaluation of the instruments calibration and the successful completion of the most recent quarterly IST.

<u>Analysis</u>: The inspectors determined the failure to perform comprehensive pump testing of CS pump P-54A in accordance with the code of record was contrary to Technical Specification 5.5.7., "Inservice Testing," and a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the mitigating system cornerstone attribute of equipment performance.

The PD adversely affected the associated cornerstone objective to ensure the availability, reliability, and capability of CS pump P-54A to respond to initiating events to prevent undesirable consequences. Specifically, failing to perform testing as required could result in the degradation of the equipment being undetected.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04," "Initial Characterization of Findings." Specifically, the inspectors used IMC 0609 Appendix A "SDP for Findings At-Power" Exhibit 2, "Mitigating Systems Screening Questions". The finding was a deficiency affecting the design or qualification of a mitigating structure system or component (SSC) but the SSC maintained its operability. This was based on the licensee's evaluation of recent pump quarterly testing, the risk evaluation performed by the licensee, and allowance to defer performing the surveillance in according with the Technical Specifications. As a result, the finding screened as having a very low safety significance, i.e. Green.

The inspectors determined the findings had a cross-cutting aspect in the area of Problem Identification and Resolution, Evaluation, because the licensee failed to thoroughly evaluate the issue to ensure that resolutions address causes and extents of conditions commensurate with their safety significance. Specifically, the licensee failed to identify that the out of calibration instrument would result in an invalid IST results, which would require rerunning the test per the code. [P.2]

Enforcement: Palisades Technical Specifications 5.5, "Programs and Manuals," states: "The following programs shall be established, implemented, and maintained," Technical Specifications 5.5.7, Inservice Testing Program, states: "This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components." Additionally, Site Engineering Program SEP-PLP-IST-102, "Inservice Testing of Selected Safety-Related Pumps" provides the details for performing Inservice Testing at Palisades and lists ASME OMb Code-2001, through ASME OMb Code-2003 Addenda, as the code of record for Palisades.

Contrary to the above, on February 25, 2014, the licensee failed to rerun a comprehensive pump test of Containment Spray pump P-54A in accordance with ASME code. Specifically, the licensee failed to meet the requirements of Section ISTB-6300, "Systematic Error," which requires licensees, after correcting the error, to rerun tests when the test shows the measured parameter falls outside the acceptable range as a result of a systematic error such as inaccurate instrumentation. The licensee entered this issue into their Corrective Action Program and evaluated the operability of the component in accordance with their process. Based on previously successful IST quarterly test the licensee determined the component remained operable. Because this violation was of very low safety significance, and it was entered into the licensee's Corrective Action Program as CR-PLP-2014-04881, this violation is being treated as a Green NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-06, Failure to Perform Comprehensive Pump Testing of Containment Spray Pump P-54A in accordance with the Inservice Testing Program.]

(6) Failure to Correctly Translate Valve Leakage Limits into Test Procedure

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for the licensee's failure to correctly translate design valve leakage limits into test procedures. Specifically, the acceptance criterion for ECCS/CS recirculation isolation valves CV-3027 and CV-3056 had not been correctly adjusted to account for the higher differential pressure associated with ECCS operation under post-accident conditions.

Description: The inspectors reviewed the leak test procedure and acceptance criteria associated with ECCS/CS recirculation isolation valves CV-3027 and CV-3056. These valves were installed in series in the common minimum flow recirculation line from the ECCS and CS pumps to the SIRW tank. During post-accident operation, these valves remain open during the ECCS injection phase to provide minimum flow protection for the low pressure safety injection (LPSI) and high pressure safety injection (HPSI) pumps. During the recirculation phase of post-accident operation, these valves are required to be closed to limit the release of radiological material from the SIRW tank vent.

The inspectors observed the design basis alternate source term analysis, NAI-1149-014, Revision 4 (dated 10/15/07), was based on a maximum allowable recirculation line leakage of 0.00625 gpm (23.7 ml/min). Test procedure RO-119, Inservice Testing of Engineered Safeguards Valves CV-3027 and CV-3056, Revision 14 included a leakage acceptance criterion of 25.1 ml/min. The inspectors also observed test Procedure RO-119 was performed with a LPSI pump in operation. The acceptance criterion had not been correctly adjusted to account for the higher differential pressure associated with HPSI pump operation under post-accident conditions. The licensee performed an informal evaluation and determined that the correct leakage acceptance criterion would be 6 ml/min. Condition Report CR-PLP-2014-04681 was initiated to address this issue.

The licensee determined that past leak test results for both CV-3027 and CV-3056 had exceeded the corrected acceptance criterion of 6 ml/min. However, the most recent leak test results for both CV-3027 and CV-3056 were less than 6 ml/min. The licensee also reviewed the results in the context of total containment leakage and verified the past leak test results did not exceed operability limits.

Analysis: The inspectors determined the failure to correctly translate design valve leakage limits into test procedures was contrary to 10 CFR Part 50, Appendix B, Criterion III, “Design Control,” and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of Design Control and it adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, measured leakage exceeded analyzed dose calculations.

The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter (IMC) 0609, “Significance Determination Process,” Attachment 0609.04, “Initial Characterization of Findings,” issued on June 19, 2012. Because the finding impacted the Barriers Cornerstone, the inspectors screened the finding through IMC 0609 Appendix A, “The Significance Determination Process for Findings At-Power,” issued June 19, 2012, using Exhibit 3, “Barrier Integrity Screening Questions.” The finding screened as of very low safety significance (Green) because the inspectors answered “No” to all of the screening questions in Subsection B, “Reactor Containment,” of Exhibit 3. Specifically, the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding did not have an associated cross-cutting aspect because it was not representative of the licensee’s present performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in §50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions.

Contrary to the above, as of September 25, 2014, the licensee failed to correctly translated design leakage limits into the test procedure for valves CV-3027 and CV-3056. Because this violation was of very low safety significance and was entered into the licensee's CAP as CR-PLP-2014-04681, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-07, "Failure to Correctly Translate Valve Leakage Limits into Test Procedure."

(7) Failure to Identify Non-Safety-Related Sub-Components Improperly Supplied with Safety-Related Valves

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," for the licensee's failure to identify non-safety-related sub-components improperly supplied with safety-related valves. Specifically, ECCS/CS recirculation isolation valves CV-3027 and CV-3056, which were installed in 2007, were supplied with non-safety-related sub-components. These components were identified as non-safety- related on the vendor drawings. In addition, the licensee later installed a section of non-safety-related tubing on valve CV-3027 based on the incorrect vendor drawing.

Description: The inspectors reviewed the vendor drawings for safety-related ECCS/CS recirculation isolation valves CV-3027 and CV-3056, which were installed in 2007. Drawings VEN-M241BC, Sheet 33, Revision 0 and VEN-M241BC, Sheet 35, Revision 0 listed the sub-components associated with these valves and identified the quality classification of each sub-component. The inspection team noted that some of the sub-components appeared to be incorrectly classified as non-safety-related. The licensee investigated and determined the drawings were incorrect and that some of these components were actually supplied as safety-related; however, several cap screws were inappropriately supplied as non-safety-related. Condition report CR-PLP-2014-04755 was initiated to correct the drawings and replace the non-safety-related cap screws with safety-related material.

The licensee performed additional investigation and determined a section of non-safety-related tubing had been installed on valve CV-3027 during maintenance activities; this error was based on the incorrect vendor drawing. Condition report CR-PLP-2014-04815 was initiated to correct the drawings and replace the tubing with safety-related material. The licensee performed an evaluation and determined both valves were operable in their current condition.

Analysis: The inspectors determined the failure to identify non-safety-related sub-components improperly supplied with safety-related valves was contrary to 10 CFR Part 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of Design Control and it adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee failed to verify the sub-components of valves CV-3027 and CV-3056 met their design requirements.

The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," issued on June 19, 2012. Because the finding impacted the Barriers Cornerstone, the inspectors screened the finding through IMC 0609 Appendix A, "The Significance Determination Process for Findings At-Power," issued June 19, 2012, using Exhibit 3, "Barrier Integrity Screening Questions." The finding screened as of very low safety significance (Green) because the inspectors answered "No" to all of the screening questions in Subsection B, "Reactor Containment," of Exhibit 3. Specifically, the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment. The inspectors determined this finding did not have an associated cross-cutting aspect because it was not representative of current performance.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion VII, "Control of Purchased Material, Equipment, and Services," requires, in part, that measures shall be established to assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents.

Contrary to the above, as of September 25, 2007 and October 8, 2007, when these valves were installed, the licensee failed to assure that purchased material conformed to the procurement documents. Because this violation was of very low safety significance and was entered into the licensee's CAP as CR-PLP-2014-04755 and CR-PLP-2014-04815, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-08, "Failure to Identify Non-Safety-Related Sub-Components Improperly Supplied with Safety-Related Valves."]

(8) Failure to Establish an Adequate Test Program for the Shutdown Cooling Heat Exchangers

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," for the licensee's failure to establish an adequate test program for the Shutdown Cooling (SDC) Heat Exchangers (HXs) to demonstrate they can perform as designed. Specifically, the licensee failed to take actions to ensure the SDC HXs' heat transfer capability met its design bases, as assumed in design bases calculations.

Description: The SDC HXs were designed for use during primary system cooldown, refueling, and emergency plant operation. During plant cooldowns and refueling outages, the HXs remove decay heat by directly cooling the primary coolant system. During emergency plant operations, the HXs form part of the containment cooling system by cooling the containment spray water. In addition, during the recirculation phase of a loss of coolant accident (LOCA), the SDC HXs cool the containment sump water that is injected by the High Pressure Safety Injection (HPSI) pumps to provide long term core cooling.

The SDC HXs are original plant equipment that has been intermittently operated since the beginning of plant operations in 1971. In over 40 years, they have never been inspected, contrary to vendor recommendations. Section E-4.1 of the vendor manual states that, "at intervals as experience indicates, an examination should be made of the interior and exterior condition of all tubes." Since the licensee has never inspected the interior of the SDC HXs, they have not established a frequency for performing the HX inspections as recommended by the vendor manual.

In addition, the SDC HXs have never been successfully thermal performance tested. In 1992, the licensee identified the need to thermal performance test the SDC HXs to verify they could perform as designed. However, because they were successfully performing their SDC function and because both water streams flowing through the HXs had chemistry controls in place, the licensee decided it was not necessary to perform the test. In 1996, the licensee completed a thermal performance test, but the analysis of the test results concluded the test did not achieve its goal of recording useful data. The test has never been repeated, and the licensee instead continues to credit the SDC function and the chemistry controls in place to provide assurance that the SDC HXs can perform as designed.

In 1989 the NRC issued Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment." Although the GL was mostly related to open cycle cooling water systems, it did contain the following guidance related to closed cycle cooling water systems:

> "Operating experience and studies indicate that closed-cycle service water systems, such as component cooling water systems, have the potential for significant fouling as a consequence of aging-related in-leakage and erosion or corrosion. The need for testing of closed-cycle system heat exchangers has not been considered necessary because of the assumed high quality of existing chemistry control programs. *If the adequacy of these chemistry control programs cannot be confirmed over the total operating history of the plant* [emphasis added] or if during the conduct of the total testing program any unexplained downward trend in heat exchanger performance is identified that cannot be remedied by maintenance of an open-cycle system, it may be necessary to selectively extend the test program and the routine inspection and maintenance program addressed in Action III, below, to the attached closed-cycle systems."

Therefore, the NRC recognized that testing of closed cycle system heat exchangers was not necessary if high quality chemistry control programs were in place. However, this was dependent on whether or not the adequacy of the chemistry control programs could be confirmed over the total operating history of the plant. Although Palisades has chemistry controls in place for both water streams (Primary Coolant System and Component Cooling Water) flowing through the SDC HXs, the licensee had not verified the effectiveness of these chemistry controls in preventing degradation of the SDC HXs heat transfer capability. The chemistry controls, in themselves, are not enough to demonstrate the SDC HXs can perform as assumed in design bases calculations.

The licensee also credits the SDC function of the HXs to provide justification that they can perform as designed. However, during SDC, the water temperatures and flow rates through the HX are different than those during emergency plant operation, and the licensee does not monitor or trend the SDC performance of the HXs to detect degradation. Therefore, although the HXs are successfully performing the routine SDC function, the licensee does not gather nor analyze sufficient HX performance information to demonstrate the HXs can perform as assumed in design bases calculations.

The issue was entered into the licensee's Corrective Action Program as CR-PLP-2014-04912 to review the adequacy of testing performed to ensure the SDC HXs remain qualified to the capability assumed during design bases events. Based upon the available tube plugging margin and recent history of chemistry controls, the inspectors gained reasonable assurance of operability.

Analysis: The inspectors determined the failure to establish an adequate test program for the SDC HXs to demonstrate they could perform as designed was contrary to 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," and was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Barrier Integrity Cornerstone attribute of Design Control and it adversely affected the associated cornerstone objective to provide reasonable assurance that containment could protect the public from radionuclide releases caused by accidents or events. Specifically, the licensee failed to verify that the SDC HXs heat transfer capability met their design bases, as assumed in design bases calculations, to limit containment temperatures and pressures during an event.

The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," issued on June 19, 2012. Because the finding impacted the Barriers Cornerstone, the inspectors screened the finding through IMC 0609 Appendix A, "The Significance Determination Process for Findings At-Power," issued June 19, 2012, using Exhibit 3, "Barrier Integrity Screening Questions." The finding screened as of very low safety significance (Green) because the inspectors answered "No" to all of the screening questions in Subsection B, "Reactor Containment," of Exhibit 3. Specifically, the finding did not represent an actual open pathway in the physical integrity of reactor containment, containment isolation system, or heat removal components and did not involve an actual reduction in function of hydrogen igniters in the reactor containment.

The inspectors determined this finding had an associated cross-cutting aspect, Conservative Bias, in the Human Performance cross-cutting area. Specifically, on several occasions and as recently as October 2012 when the licensee identified the need to perform testing and/or inspection of the SDC HXs, the licensee did not take actions because they did not believe any regulatory requirements or technical issues existed that required the testing and/or inspections. [H.14]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure all testing required to demonstrate structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents.

Contrary to the above, as of November 4, 2014, the licensee failed to establish an adequate test program for the SDC HXs to demonstrate they could perform as designed. Specifically, the licensee failed to take actions to ensure the heat transfer capability of the SDC HXs met their design bases, as assumed in design bases calculations.

Because this violation was of very low safety significance and was entered into the licensee's CAP as CR-PLP-2014-04912, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-09, "Failure to Establish an Adequate Test Program for the Shutdown Cooling Heat Exchangers"]

(9) Failure To Include The Degraded Voltage Channel Time Delay In TS Surveillance Requirement 3.3.5.2.a

Introduction: The inspectors identified a finding having very low safety significance and an associated Non-Cited Violation (NCV) of 10 CFR Part 50.36(c)(3), "Surveillance Requirements," for the failure to ensure the channel time delay for the degraded-voltage monitor was included in TS SR 3.3.5.2.a. Specifically, the licensee had only included the timing requirements for the voltage sensing relay in the TS SR; failing to include the required time delay after the voltage sensing relay trips before the preferred source of power is isolated and 1E electrical loads transferred to the stand-by EDGs.

Description: The inspectors noted TS SR 3.3.5.2, Revision 189, requires the licensee to "Perform a CHANNEL CALIBRATION on each Loss of Voltage and Degraded Voltage channel with setpoints…:" Part a. of SR 3.3.5.2 gives the degraded voltage function setpoint but only gives the time delay associated with the sensing element of the degraded voltage monitor. However, a channel of the degraded voltage monitor contains both the voltage sensing relay and a nominal 6-second delay timer which has to time out before the trip relay actuates and off-site power supply breaker is opened.

The inspectors reviewed an NRC letter to the licensee dated December 22, 1981, with the Subject: SEP Topic VIII-1.A, "Potential Equipment Failures Associated With Degraded Grid Voltage." Enclosure 1 to this letter was the NRC staff's Safety Evaluation Report (SER) on degraded grid voltage protection for the Class 1E System. The SER conclusions were based on a detailed review and technical evaluation of the licensee's proposed modifications and changes to the Technical Specifications. This SER referenced EG&G Interim Report No. EGG-EA-5321, Revision 3, Degraded Grid Protection for Class 1E Power Systems, Palisades Plant, Docket No. 50-255, TAC No. 10043, Section 3.3, "Discussion," Paragraph 3, which states, in part, "The time delay selected shall be based on the following conditions: a; The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the FSAR accident analysis, and b; The time delay shall minimize the effect of short duration disturbances from reducing the unavailability of the offsite power source(s)." Paragraph 4 states, "The voltage monitors shall automatically initiate the disconnection of offsite power sources whenever the voltage setpoint and time-delay limits have been exceeded." Paragraph 6 states, "The Technical Specifications shall include limiting conditions for operations, surveillance requirements, setpoints with minimum and maximum limits, and allowable values for the second-level voltage protection monitors." (Note that the degraded voltage setpoint is also known as the second-level voltage relay setpoint.) The total time delay evaluated by EGG was a nominal 6 seconds added to the time for the voltage relay to activate. From these statements, it was clear that a degraded voltage monitor channel consisted of both the voltage sensing relay and the delay timer.

The licensee did not make a change to the custom TS but decided to enter the changes when implementing the standardized Combustion Engineering TS. The inspectors reviewed the Combustion Engineering STS Bases 3.3.7.3, which states, in part, "SR 3.3.7.3 is the performance of a CHANNEL CALIBRATION. The CHANNEL CALIBRATION verifies the accuracy of each component within the instrument channel." Additionally, NRC Regulatory Issue Summary 2011-12, Revision 1, "Adequacy of Station Electric Distribution System Voltages," again states, in part, "The voltage monitors (or DVRs) shall automatically initiate the disconnection of offsite power source(s) whenever the voltage and time delay limits have been exceeded," and "The Technical

Specifications shall include … Allowable values for second-level voltage protection DVRs." After the inspectors communicated this concern as a non-conservative TS, the licensee initiated condition report CR-PLP-2014-04903, "TS SR 3.3.5.2 is Non-Conservative," and implemented administrative controls in accordance with Administrative Letter 98-10 for SR 3.3.5.2 degraded voltage time. The operability evaluation found the diesel generator under voltage start circuitry to be operable DNC [degraded non-conforming] because the total time delay had been checked and calibrated to within acceptable values.

Analysis: The inspectors determined the failure to incorporate the total time delay in the 2400 V Class 1E bus degraded voltage monitor channel trip setpoints in the Technical Specification surveillance requirements was contrary to 10 CFR Part 50.36(c)(3), and was a performance deficiency. The performance deficiency was determined to be more than minor because if left uncorrected, it would have the potential to lead to more significant safety concern. Specifically, by not incorporating the total time delay requirements into the Technical Specifications, the time could be changed without going through the TS change process, possibly leading to spurious trips of offsite power sources or possibly exceeding the accident analysis time is the FSAR.

In accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," Table 2 the inspectors determined the finding affected the Initiating Events cornerstone due to the potential to initiate a loss of offsite power. As a result, the inspectors determined the finding could be evaluated using Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions." The inspectors determined the finding was of very low safety significance (Green) because it did not cause a reactor trip and the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition.

The inspectors did not identify a cross-cutting aspect associated with this finding because the finding was not representative of the licensee's present performance.

Enforcement: Title10 CFR Part 50.36(c) states, Technical Specifications will include minimum surveillance requirements. Title10 CFR Part 50.36(c)(3) "Surveillance Requirements," states, in part, "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

Contrary to the above, as of October 9, 2014, the licensee failed to include within TS SR the total time delay for all components in the degraded voltage, which is required to maintain facility operation within safety limits. Specifically, the licensee failed to incorporate the total time delay for all of the components in a channel as required in a CHANNEL CALIBRATION. Because this violation was of very low safety significance, and it was entered into the licensee's Corrective Action Program as CR-PLP-2014-04903, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-10, Failure To Include The Degraded Voltage Channel Time Delay In TS Surveillance Requirement 3.3.5.2.a].

(10) Lack of Analysis for Electrical Containment Penetration Protection

Introduction: The inspectors identified an unresolved item (URI) regarding lack of an analysis to demonstrate that circuit breakers and fuses provide adequate protection against short circuits and overloads for electrical containment penetrations, as discussed in Regulatory Guide 1.63. Resolution of this issue will be based on clarification of Palisade's licensing basis by NRC staff.

Description: As part of the review of power supplies to components inside containment, the inspectors requested to review the analysis that demonstrates protection of the electrical penetrations against short circuits and overloads. The licensee responded that such an analysis does not exist, and stated their position that it is not required by their design and licensing bases.

Electrical protection of containment penetrations was the subject of the Palisades Systematic Evaluation Program (SEP) Topic VIII-4. A letter from Dennis M. Crutchfield, NRC, to David P. Hoffman, Consumers Power Company, "SEP Topic VIII-4, Electrical Penetrations of Reactor Containment," dated March 26, 1981, (ADAMS Accession No. ML8104080152) included an enclosure entitled "Position on Protection of Containment Electrical Penetrations against Failures Caused by Fault and Overload Currents for SEP Plants." This position document states: "…the staff requires compliance with the recommendations of Regulatory Guide 1.63 or an acceptable alternative method. For each containment electrical penetration, the protective systems provide primary and backup protection devices to prevent a single failure in conjunction with a circuit overload from impairing containment integrity."

The licensee responded in a letter from Robert A. Vincent, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Electrical Penetrations of Reactor Containment," dated June 15, 1981, (ADAMS Accession No. ML8106180170), in which they stated: "The secondary (backup) interrupt devices (…) would fail to trip prior to the penetration reaching its limiting temperature of 302° C with the postulated combination of faults and failure of the primary interrupters." The licensee committed to perform more detailed evaluations of the capabilities of the protective devices, as well as "An evaluation of the adequacy of the Palisades Plant overcurrent protection surveillance testing program." In the following subsequent letters, the licensee reported on the progress of their further evaluations: Letter from Robert A. Vincent, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Electrical Penetrations of the Reactor Containment," dated November 16, 1981, (ADAMS Accession No. ML8111200805); Letter from Kerry A. Toner, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Electrical Penetrations of the Reactor Containment," dated October 12, 1982, (ADAMS Accession No. ML8210190459); and letter from Kerry A. Toner, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Status Update of Program to Evaluate the Adequacy of Penetration Protection from Overload and Short-Circuit Conditions," dated February 11, 1983, (ADAMS Accession No. ML8302240273).

The NRC issued their Integrated Plant Safety Assessment Report on this SEP Topic in Letter from Thomas V. Wambach, NRC, to David J. VandeWalle, Consumers Power Company, "Integrated Plant Safety Assessment Report (IPSAR) Section 4.26, Electrical Penetrations of Reactor Containment – Palisades Plant," dated June 10, 1983, (ADAMS Accession No. ML8306160396). This IPSAR stated, "The staff has evaluated this issue for other plants… and concluded that no further action was required for these plants. Based upon the information contained in the licensee's letters dated June 15, 1981, October 12, 1982, and February 11, 1983, the staff concludes that the design of the Palisades electrical penetrations are similar to other SEP plants, that the probability of electrical failure is low and that any leakage path due to penetration failure would be small. Therefore, we consider this issue to have been completed satisfactorily and further action by the licensee is not required." This conclusion was reiterated in NUREG 0820, Supplement 1, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Palisades Plant, Final Report," dated November 1983, (ADAMS Accession No. ML8311290133).

The inspectors requested that the licensee discuss their lack of an analysis for electrical penetration protection in light of Palisades FSAR Section 8.5.1.2, which states: "10 CFR Part 50, General Design Criterion 50, as implemented by Regulatory Guide 1.63 and IEEE Standard 317-1972, requires that electrical penetrations be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the calculated pressure, temperature and other environmental conditions resulting from any Loss of Coolant Accident (LOCA)." The licensee responded by initiating corrective action CR-PLP-2014-04450, which states the licensee's position that Palisades is not committed to "the electrical circuit protection requirements of Regulatory Guide 1.63."

Due to complexity of establishing the appropriate design and licensing bases for this issue, the concern will be resolved using the NRC's Task Interface Agreement (TIA) process. Pending resolution, this item will be tracked as an unresolved item. [URI 05000255/2014008-05, Lack of Analysis for Electrical Containment Penetration Protection].

(11) <u>Classification of CCW Piping and Components Inside the Reactor Containment Building</u>

<u>Introduction</u>: The inspectors identified an unresolved issue (URI) regarding the Inservice Inspection (ISI) classification of component cooling water (CCW) piping and components inside the reactor containment building. This piping is currently classified as non-safety-related. Resolution of this issue will be based on clarification of Palisade's licensing basis by NRC staff.

<u>Description</u>: The inspectors reviewed SEP-ISI-PLP-002, ASME Code Boundaries for ASME Section XI Inservice Inspection Program, Revision 1. The purpose of this document was to establish classifications to ensure the proper scope of ASME Section XI system pressure tests, ASME Section V nondestructive examinations, ASME Section XI inspection of component supports, and ASME Section XI repairs, replacements and modifications. This program document stated that guidance for Class 2, 3, and non-safety-related components is found in Regulatory Guide 1.26. The program referred to a set of color-coded P&IDs.

The inspectors noted the CCW piping and components inside the reactor containment building were identified as non-safety-related on the color-coded P&IDs. The team also observed that Attachment 1 of SEP-ISI-PLP-002 (page 45 of 75) identified the piping from the containment penetrations to the single containment isolation valve outside containment as class 2 (check valve CK-CC910 for penetration MZ-14, air-operated valve CV-0911 for penetration MZ-15). This attachment referred to note H, which stated, in part:

> *All containment penetration assemblies in Class 2, Class 3, and non-class piping will be considered ASME Class 2 out to the second isolation valve where applicable. In some cases, there is single isolation (e.g., MSS, FWS)… Where a Class 3 system penetrates containment, that portion will be considered Class 2 and treated as such (ie, ASME Section XI Interpretation BC84-603). The basis for classification of containment penetrations is contained in EGAD-EP-12, "Mechanical Containment Penetrations Basis Program".*

The current Mechanical Containment Penetrations Basis document was SEP-APJ-PLP-101, Revision 0. This document included a similar description for CCW penetrations MZ-14 and MZ-15. It stated that FSAR Table 5.8-4 classifies these penetrations as class C1. The FSAR table also identified these penetrations as Class C1 and indicated that they are not subject to 10 CFR Part 50, Appendix J testing requirements. The FSAR Section 6.7 included a description of Class C1 penetrations, it stated, in part:

> *Penetrations in this class include those systems that are not connected to either the containment atmosphere or to the Primary Coolant System and are normally open or may be opened during power operation. These lines are protected from missiles originating inside the containment and the lines themselves form the boundary of the containment. One remote manually operated valve, locked closed manual valve or automatic isolation valve is provided in each line. Check valves are considered automatic.*

In SEP-APJ-PLP-101 also stated, that both Penetrations MZ-14 and MZ-15 were originally classified as C2 (closed system outside containment per FSAR Section 6.7), until FSAR Revision 22. These were reclassified by SDR-99-0884, dated July 22, 1999. The basis for these changes included an evaluation that determined CCW piping inside containment would not be damaged by internal missiles (as addressed by NRC's evaluation of SEP Topic III-4,c, dated September 21, 1981), and the classification of CCW as a closed system inside containment. The associated 10 CFR 50.59 evaluation determined that NRC approval was not required for these changes.

The inspection team raised questions regarding the licensing basis classification of the CCW system inside containment:

1. FSAR Section 6.9 stated that Regulatory Guide 1.26 was used to select ASME Classes 2 and 3 systems and components for coverage by the inspection plan. Regulatory Guide 1.26, stated in part:

   > *The Quality Group C standards given in Table 1 of this guide should be applied to water-, steam-, and radioactive-waste-containing pressure vessels; heat exchangers (other than turbines and condensers); storage tanks; piping; pumps; and valves that are not part of the reactor coolant pressure boundary or included in Quality Group B but part of the following:*
   >
   > *…(b) cooling water and seal water systems or portions of those systems important to safety that are designed for the functioning of components and systems important to safety, such as reactor coolant pumps, diesels, and the control room…*

   The classification of the CCW system inside containment did not appear to be consistent with the guidance of Regulatory Guide 1.26. In addition, industry guidance provided by ANSI/ANS 51.1 – 1983 indicated that closed systems inside containment (with single active isolation valves) should be class 2 or class 3.

2. FSAR Section 6.7 addressed class C1 containment penetrations stating, "These lines are protected from missiles originating inside the containment and the lines themselves form the boundary of the containment." The inspectors questioned whether it was appropriate to classify a portion of the containment boundary as non-safety-related for the purpose of inspection, testing, and repairs.

The licensee has stated that the CCW system classification is in accordance with their current licensing basis. This position is primarily based on the NRC's evaluation of SEP Topic III-4,c, dated September 21, 1981, the CCW system being protected from internal missiles, and seismic events not being postulated to occur coincidently with a LOCA.

Resolution of this issue will be based on clarification of Palisade's licensing basis. Pending resolution, this item will be tracked as an URI. [URI 05000255/2014008-11, Classification of CCW Piping and Components Inside the Reactor Containment Building.]

(12) <u>Component Cooling Water System Licensing Bases</u>

<u>Introduction</u>: The inspectors identified an Unresolved Item (URI) regarding the licensing bases for the Component Cooling Water (CCW) system. Specifically, the inspectors require clarification as to what failures of the CCW system the licensee needs to postulate and evaluate. The NRC will conduct further inspection to determine when these changes to the licensing bases occurred.

<u>Description</u>: As part of the 2014 Component Design Bases Inspection (CDBI), the inspectors selected CCW pump P-52B and relief valve RV-0956 for review. Both of these components were part of the CCW system. The CCW system was designed as a closed cycle system, where both trains share a common suction and common discharge header. This means that although there were redundant pumps and heat exchangers, the system's piping was not designed to be redundant and a single pipe break or failure of the pressure boundary could result in the complete loss of CCW. One of CCW's safety functions was to transfer heat from the reactor and containment (post-Design Bases Events/Accidents) to the ultimate heat sink. Another important safety function for CCW was to provide cooling to the Engineered Safeguard Systems' (ESS) and containment spray (CS) pumps. Per the licensee's design bases, cooling to the ESS pumps was required to maintain their operability.

When reviewing the licensing bases for the plant, it was not clear what type of failures needed to postulated for the CCW system under post-accident conditions. The licensee's position was postulating a passive failure of CCW concurrent with a design bases accident (DBA) was not within their licensing bases. The licensee's position was that no active single failure, according to their definition in FSAR Section 1.4.16, would render CCW inoperable. They also considered a postulated failure of the non-safety-related portion of the CCW system inside containment as beyond design bases, except as result of a seismic event which was not postulated to occur in conjunction with an accident.

Currently, the licensee credits post-accident heat being removed from containment by a combination of containment air coolers (CAC) and the containment spray (CS) system. The CAC are supplied by service water and are independent of the CCW system. Per the current design, the licensee needs either two CS pumps or one CS pump and three CACs. Both alternatives require the CCW system to remove heat from the CS system. However, the original design took credit for the CS and the CAC as independent and redundant in their capability to remove heat from the containment. In other words, originally the licensee needed either two CS pumps or three CACs. Additionally, the original design allowed for the capability to swap cooling water to the ESS pumps from CCW to service water remotely from the main control room (MCR). Both of these design flexibilities have been either lost or eliminated due to subsequent design changes.

The inspectors noted the agency staff had previously evaluated the susceptibility of CCW to loss of function following certain assumed CCW pipe breaks during the Systematic Evaluation Program(SEP). This was documented on SEP Topic IX-3, Station Service and Cooling Water Systems Palisades, February 22, 1982. The agency staff had concluded the CCW design was not in conformance with GDC 44, regarding capability and redundancy of essential functions of the system. However, the staff noted the essential functions of CCW could be performed by other systems under all operating conditions. The SEP evaluation explicitly addressed a passive failure of the CCW system under post-accident conditions and concluded that the CACs would be capable of removing heat from containment.

The inspectors were concerned that if the CCW system became inoperable as the result of non-safety-related component failures, the plant would no longer have the redundant capability to remove heat from the containment during a DBA, or provide alternate cooling to the ESS pumps from the MCR. In addition, the inspectors needed to clarify the licensing bases regarding a postulated loss of CCW concurrent with a design bases accident.

This issue is unresolved pending further inspection to determine when these changes to the licensing bases occurred. [URI 05000255/2014008-12, Component Cooling Water System Licensing Bases]

.4 Operating Experience

a. Inspection Scope

The inspectors reviewed six operating experience issues to ensure that NRC generic concerns had been adequately evaluated and addressed by the licensee. The operating experience issues listed below were reviewed as part of this inspection:

- Information Notice 1988-45, "Problems in Protective Relay and Circuit Breaker Coordination;"
- Information Notice 2010-09, "Importance of Understanding Circuit Breaker Control Panel Indications";
- Information Notice 2010- 11, "Potential for Steam Voiding Causing Residual Heat Removal System Inoperability";
- Regulatory Issue Summary 2013-05, "NRC Position On The Relationship Between General Design Criteria and Technical Specification Operability";
- NRC-21-2013-60-00 and -01, "K-Line Circuit Breaker Primary Close Latch"; and
- IER-L4-13-54, "Unprotected DC Ammeters Result in Unanalyzed Condition."

b. Findings

No findings of significance were identified.

.5 Modifications

a. Inspection Scope

The inspectors reviewed one permanent plant modification related to selected risk significant components to verify that the design bases, licensing bases, and performance capability of the components had not been degraded through modifications. The modification listed below was reviewed as part of this inspection effort:

- FC-718,RAS Logic Seal-In Circuit Addition;

b. Findings

No findings of significance were identified.

.6 Operating Procedure Accident Scenarios

a. Inspection Scope

The inspectors performed a detailed review of the operator actions and the procedures listed below associated with the selected components and associated with risk important operator actions. For the procedures listed, simulator scenarios were observed as applicable, and in-plant actions were walked down with a non-licensed operator or a licensed operator as appropriate. These activities were performed to determine whether there was sufficient information to perform the procedure, whether the steps could reasonably be performed in the available time, and whether the necessary tools and equipment were available. The procedures were compared to FSAR and design assumptions. In addition, the procedures were reviewed to ensure the procedure steps would accomplish the desired results.

The following Time Critical Operator Actions (TCAs) and accidents were reviewed:

- TCA 1: "Enable Closure of the ESS recirculation valves on RAS,"
- TCA 6: "Align hot leg injection,"
- TCA 44: "Switch CR HVAC to Emergency Mode,"
- TCA 47: "ESS Suction Header Cross-Tie,"
- Large Break Loss of Coolant Accident (LBLOCA),
- Mode 4 Loss of Coolant Accident (LOCA).

The following procedures were reviewed:

- Administrative Procedure TCA, "Control of Time Critical Operator Actions," Revision 3;
- Emergency Operating Procedure Basis EOP TCA, "EOP Time Critical Operator Action Basis," Revision 0;
- Emergency Operating Procedure EOP-4.0, "Loss of Coolant Accident Recovery," Revision 23;
- Emergency Operating Procedure Supplement 42, "Pre and Post RAS Actions," Revision 7
- Abnormal Operating Procedure AOP-30, Loss of Shutdown Cooling," Revision 1.

b. Findings

No findings of significance 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 inspectors reviewed a sample of the selected component problems that were identified by the licensee and entered into the Corrective Action Program. The inspectors 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 Corrective Action Program. The specific corrective action documents that were sampled and reviewed by the inspectors are listed in the attachment to this report.

The inspectors also selected six issues that were identified during previous CDBIs to verify the concern was adequately evaluated and corrective actions were identified and implemented to resolve the concern, as necessary. The following issues were reviewed:

- NCV 05000255/2006009-08, "Emergency Diesel Generator Fuel Transfer System Temperature Ratings";
- NCV 05000255/2006009-10, "Potential for Safety Injection and Refueling Water Tank Level Switch Setpoints to be Outside Technical Specification Limit";
- NCV 05000255/2008009-01, "Inadequate Analysis of Emergency Diesel Generator 1–2 Loading During Design Basis Events";
- NCV 05000255/2008009-02, Failure to Establish Correct Technical Specification Limits";
- NCV 05000255/2011009-01, Failure to Adequately Evaluate the Enclosure Installed Over the 1F/1G Buses"; and
- NCV 05000255/2011009-03, "Procedures Were Not Appropriate To Address Gas Accumulation Issues"

b. Findings

(1) Non-Conservative Surveillance for Emergency Diesel Generator Largest Load Reject Test

Introduction: A finding of very low safety significance and associated Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was identified by the inspectors for the licensee's failure to have adequate acceptance criteria in the emergency diesel generator surveillance procedures. Specifically, the licensee failed to ensure the surveillance test procedures for the emergency diesel generator largest load reject test bounded the power demand of the largest load, as required by Technical Specification SR 3.8.1.5 based upon full load reject testing.

Description: Palisades Technical Specification Surveillance Requirement SR 3.8.1.5 requires verification of the emergency diesel generator response following rejection of "a load greater than or equal to its associated single largest post-accident load." On June 20, 2013, the licensee issued Technical Specification Surveillance Procedure RE-132, "Diesel Generator 1-2 Load Reject," Revision 6. This procedure specifies, in Step 5.3.11, that the emergency diesel generator be loaded to "325 to 375 kw" in preparation for the load reject test. On August 6, 2014, the licensee issued calculation EA-ELEC-LDTAB-005, "Emergency Diesel Generators 1-1 and 1-2 Steady State Loadings," Revision 10, which includes documentation of the magnitudes of the emergency diesel generator loads. The inspectors asked the licensee to compare the calculation data with the procedure criterion for a minimum test load of 325kW. The licensee concluded that, according to the calculation, the largest single emergency diesel generator load is High Pressure Safety Injection Pump P-66A, which has a power demand of 360.25kW. Thus, surveillance Procedure RE-132 allows an emergency diesel generator load test value (325kW) that does not bound the magnitude of the generator's single largest load (360.25kW).

The licensee entered this issue into their Corrective Action Program as CR-PLP-2014-4679 and CR-PLP-2014-4680 for Emergency Diesel Generators 1-1 and 1-2, respectively, with recommended actions to revise the surveillance procedure minimum load limit for the load reject test to bound the magnitudes of the single largest loads. The licensee also verified current operability by confirming the emergency diesel generators are capable of performing as required by SR 3.8.1.5.

Analysis: The inspectors determined the failure to have adequate acceptance criteria in the emergency diesel generator surveillance procedure was contrary to 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the surveillance procedure error could result in acceptance of test results that did not satisfy Technical Specification SR 3.8.1.5 for rejection of a load greater than or equal to the emergency diesel generator's single largest post-accident load.

The inspectors assessed this finding for significance in accordance with NRC Manual Chapter 0609, Appendix A, Exhibit 2, The Significance Determination Process (SDP) for Findings At-Power, and determined that it was of very low safety significance (Green), because the SSC maintained its operability and functionality. This finding has a cross-cutting aspect in the area of Human Performance, associated with the Resources component, because the licensee did not ensure that personnel, equipment, procedures, and other resources are adequate to assure nuclear safety by maintaining long term plant safety. [H.1]

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," requires, in part, "A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents." Contrary to the above, on June 20, 2013, the licensee failed to assure that written test procedures incorporate requirements contained in applicable design documents. Specifically, the licensee revised Technical Specification Surveillance Procedure RE-132, "Diesel Generator 1-2 Load Reject," but failed to identify that the specified kW level for the largest load rejection test did not bound the largest predicted post-accident load.

Because this violation was of very low safety significance, and it was entered into the licensee's Corrective Action Program as CR-PLP-2014-4679 and CR-PLP-2014-4680, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. [NCV 05000255/2014008-13, Non-Conservative Surveillance for Emergency Diesel Generator Largest Load Reject Test.]

4OA5 Other Activities

.1 (Closed) Unresolved Item (URI) 05000255/2012005-05: Concerns With The Methodology Used To Determine Suction Side Void Acceptance Criteria

On January 11, 2008, the NRC requested each addressee of GL 2008-01 to evaluate its emergency core cooling, decay heat removal, and containment spray systems licensing basis, design, testing, and corrective actions to ensure gas accumulation was maintained less than the amount which would challenge the operability of these systems, and take appropriate actions when conditions adverse to quality were identified. In order to determine what amount of gas could challenge the operability of the subject systems, the licensee needed to develop appropriate acceptance criteria for evaluating identified voids. As part of this effort, the licensee developed acceptance criteria for evaluating voids identified in the suction side of the subject systems' pumps.

During subsequent inspections the inspectors identified a concern with the methodology used by the licensee to develop acceptance criteria for suction side voids. At the time of the inspection the acceptance criteria was inconsistent with the 0.5-second criterion recommended by the Office of Nuclear Reactor Regulation (NRR) in TI 2515/177 Inspection Guidance (ML111660749). The NRR-recommended methodology was more conservative because it ensured there were no significant deviations exceeding the maximum recommended void fractions. However, because the licensee's methodology averaged over the entire transient duration time, it allowed void volumes that could significantly exceed the recommended void fraction when the actual duration transient time was shorter than the maximum allowable duration time specified by the recommended void fraction acceptance criteria. This issue was originally captured in the licensee's CAP as CR-HQN-2011-00853.

Since then, and as result of other void related violations, the licensee revised their procedures including their Gas Accumulation Management Program Document (EN-DC-219, Revision 3). The revised Program Document incorporates, by reference, the Acceptance Criteria of NEI 09-10, "Guidelines for Effective Prevention and Management of System Gas Accumulation." The NRC has issued a Safety Evaluation (SE) of the NEI 09-10, which discusses the Agency's position in regards to the use of NEI 09-10. In addition, based on the information discussed in the NEI 09-10 and the SE, the 0.5 second criterion discussed in the URI is no longer the recommended method proposed by NRR. Therefore, the inspectors concern is no longer valid, and no performance deficiency or violation of regulatory requirements exists. This unresolved item is closed.

4OA6 Management Meeting(s)

.1 Exit Meeting Summary

On November 4, 2014, the inspectors presented the inspection results to Mr. Vitale, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors asked the licensee if the inspectors had proprietary materials that should be returned. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

.2 Followup Exit Meeting Summary

On November 4, 2014, the inspectors conducted a follow-up exit by phone with Mr. Vitale, and other members of the licensee staff. This meeting updated the status of two concerns that were open at the time of the on-site exit. The licensee acknowledged the issues presented.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy, for being dispositioned as a Non-Cited Violation.

- Technical Specification 5.4.1.a, requires written procedures shall be established, implemented, and maintained as recommended in Regulatory Guide 1.33, Item (b), Administrative Procedures implementing the "Authorities and Responsibilities for Safe Operation and Shutdown." Procedure 4.48, Revision 3, "Control of Time Critical Operator Actions." Requires that Time Critical Operator Actions (TCAs) revalidated every two years. Contrary to the above, on July 23, 2014, the licensee found no evidence that TCA revalidation was performed since 2011. The licensee determined timing information could not be found for the following TCA validations:

    - TCA 2: Match PCS and S/G Pressure during a SGTR Event,
    - TCA 10: Align Shutdown Cooling (SDC), and
    - TCA 46: Station Battery Load Stripping.

  This issue was determined to be more than minor, because it impacted the Procedure Quality attribute of the Reactor Safety, Mitigating Systems Cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The licensee documented this issue in CR-PLP-2014-03841, dated July 23, 2014. The finding was determined to be of very low safety significance (Green) because all questions were answered "no" for Exhibit 2 – Mitigating Systems Screening Questions in Inspection Manual Chapter (IMC) 0609, Appendix A, Significance Determination Process (SDP) For Findings At-Power.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

T. Vitale, Site Vice President
T. Williams, General Manager, Plant Operations
B. Davis, Engineering Director
D. Corbin, Operations Manager
J. Borah, System Engineering Manager
K. O'Connor, Design Engineering Manager
J. Hardy, Regulatory Assurance Manager
B. Sova, Engineering Supervisor
T. Fouty, Engineering Supervisor
L. Engelke, Engineering Supervisor
K. Yeager, Engineering Supervisor
D. MacMaster, Engineering Supervisor
R. Scudder, Operations Specialist
B. Dotson, Regulatory Assurance Specialist
T. Davis, Regulatory Assurance Specialist

U.S. Nuclear Regulatory Commission

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000255/2014008-01	NCV	Failure to Ensure Engineered Safeguards Systems Aren't Adversely Affected By Air Entrainment (Section 1R21.3.b(1))
05000255/2014008-02	NCV	Undersized Supply Cables from Startup Transformer to 2400V Buses (Section 1R21.3.b(2))
05000255/2014008-03	NCV	Undersized Motors (Section 1R21.3.b(3))
05000255/2014008-04	NCV	Failure to Ensure that 480V System Voltages do not Exceed Equipment Ratings (Section 1R21.3.b(4))
05000255/2014008-05	NCV	Failure to Perform Comprehensive Pump Testing of Containment Spray Pump P-54A in accordance with the Inservice Testing Program (Section 1R21.3.b(5))
05000255/2014008-06	NCV	Failure to Correctly Translate Valve Leakage Limits into Test Procedure (Section 1R21.3.b(6))
05000255/2014008-07	NCV	Failure to Identify Non-Safety-Related Sub-Components Improperly Supplied with Safety-Related Valves (Section 1R21.3.b(7))
05000255/2014008-08	NCV	Failure to Establish an Adequate Test Program for the Shutdown Cooling Heat Exchangers (Section 1R21.3.b(8))

05000255/2014008-09	NCV	Failure To Include The Degraded Voltage Channel Time Delay In TS Surveillance Requirement 3.3.5.2.a (Section 1R21.3.b(9))
05000255/2014008-10	URI	Lack of Analysis for Electrical Containment Penetration Protection (Section 1R21.3.b(10))
05000255/2014008-11	URI	Classification of CCW Piping and Components Inside the Reactor Containment Building (Section 1R21.3.b(11))
05000255/2014008-12	URI	Component Cooling Water System Licensing Bases (Section 1R21.3.b(12))
05000255/2014008-13	NCV	Non-Conservative Surveillance for Emergency Diesel Generator Largest Load Reject Test (Section 4OA2.1.b(1))

<u>Closed</u>

05000255/2014008-01	NCV	Failure to Ensure Engineered Safeguards Systems Aren't Adversely Affected By Air Entrainment (Section 1R21.3.b(1))
05000255/2014008-02	NCV	Undersized Supply Cables from Startup Transformer to 2400V Buses (Section 1R21.3.b(2))
05000255/2014008-03	NCV	Undersized Motors (Section 1R21.3.b(3))
05000255/2014008-04	NCV	Failure to Ensure that 480V System Voltages do not Exceed Equipment Ratings (Section 1R21.3.b(4))
05000255/2014008-06	NCV	Failure to Perform Comprehensive Pump Testing of Containment Spray Pump P-54A in accordance with the Inservice Testing Program (Section 1R21.3.b(6))
05000255/2014008-07	NCV	Failure to Correctly Translate Valve Leakage Limits into Test Procedure (Section 1R21.3.b(7))
05000255/2014008-08	NCV	Failure to Identify Non-Safety-Related Sub-Components Improperly Supplied with Safety-Related Valves (Section 1R21.3.b(8))
05000255/2014008-09	NCV	Failure to Establish an Adequate Test Program for the Shutdown Cooling Heat Exchangers (Section 1R21.3.b(9))
05000255/2014008-10	NCV	Failure To Include The Degraded Voltage Channel Time Delay In TS Surveillance Requirement 3.3.5.2.a (Section 1R21.3.b(10))
05000255/2014008-13	NCV	Non-Conservative Surveillance for Emergency Diesel Generator Largest Load Reject Test (Section 4OA2.1.b(1))
05000255/2012005-05	URI	Concerns With The Methodology Used To Determine Suction Side Void Acceptance Criteria (Section 4OA5.1)

<u>Discussed</u>

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.

CALCULATIONS

Number	Description or Title	Revision
152-202	Installation of Replacement Siemens Breaker	1
152-203	Installation of Replacement Siemens Breaker	1
1D-208-150-151	Component Cooling Pump P-52B	4
1D-210-150-151	Breaker Calculation for 152-210	3
1D-210-150-151	Breaker Calculation for 152-210	3
1D-210-150-151	Breaker Calculation for 152-210	3
23-3E	LPSI Valve MO-3190 (52-2339) ESPP	3
D11/04	4160V Switchgear Bus 1F DC Supply	1
DBD 5.01	Diesel Engine and Auxiliary Systems	6
DBD 5.03	Emergency Diesel Generator Performance Criteria	8
DCP-090188-1	Range of Station Power Voltages	0
EA-AOVCAP-GATE-ESS-01	Actuator Capacity Review for Air Operated Gate Valves in ESS	1
EA-AOVT/T-ESS-02	Stem Thrust Requirements for AOVs CV-3027 and CV-3056	2
EA-AOVT/T-ESS-03	Evaluation of Thrust Requirements for AOVs CV-3029 and CV-3030	1
EA-AOV-WKLINK-02	Weak Link Calculation for AOVs CV-3027 and CV-3056	1
EA-AOV-WKLINK-06	Weak Link Calculation for AOVs CV-3029 and CV-3030 from Crane Valve	1
EA-A-PAL-90-018-01	DBA Sequencer Timing Study	3
EA-A-PAL-92-037	Emergency Diesel Generators Loadings--First Two Hours	1
EA-APR-95-001	Appendix R Safe Shutdown Equipment List and Logic Diagrams	4
EA-APR-95-002	10 CFR Part 50 Appendix R Safe Shutdown Associated Circuits Analysis and Cable Selection	1
EA-APR-95-004	10 CFR Part 50 Appendix R Safe Shutdown Associated Circuits Analysis for Common Power Supply and Common Enclosure	5
EA-APR-95-015	Diesel Generator Capability without Service Water	0
EA-CA024154-01	Containment Spray System Flow Rates and Timing During Injection Mode Using Pipe-Flo	1

EA-CA025644-01	Evaluation of the Impact of 110 Percent EDG Overload on Ambient Temperature	1
EA-CCW-DBD-89-01	Generation of Design Basis for Specific Relief Valve Sizes in the CCW System	0
EA-C-PAL-95-0877D	Evaluation of the Potential for Excessive Air Entrainment Caused by Vortexing in the SIRW tank During a LOCA	1
EA-C-PAL-97-1650A-01	Diesel Generator Hydraulic Inputs	2
EA-C-PAL-97-1650A-01	Revised Hydraulic Inputs for Emergency Diesel Generator Steady State Load	2
EA-C-PAL-01-03563-02	Containment Sump NPSH Evaluation	0
EA-EAR-97-0273	Evaluation of 4 November 1996 Shutdown Cooling Heat Exchanger Performance Test T-365	0
EA-EAR-2001-0333-01	Generation of ESS Pump Performance Curves for use with the Pipe-Flo ESS Hydraulic Model	5
EA-EC157-01	345 KV System Voltage Drop with LOCA - Replacement SUT 1- 2	0
EA-EC-235-01	Assessment of the High Pressure Air System's Capacity to Cycle Valves in the West Engineering Safeguards	0
EA-EC496-001	Evaluation of Downcomer, Floor Drain, and Sump Vent Screens for GSI-191 Resolution	1
EA-EC496-04	SFS Surface Area, Flow and Volume	1
EA-EC6432-01	Palisades Emergency Diesel Generator Fuel Oil Storage Requirements	1
EA-EC7107-01	Palisades GSI-191 Downstream Effects Evaluation of ECCS Components	2
EA-EC-8349-04	Post LOCA Boric Acid Precipitation Analysis for Palisades in Support of Replacement of TSP with NaTB as the Containment Sump Buffer Agent	0
EA-EC9600-01	Functionality of Equipment in EDG Room at an Elevated Temperature of 121F	1
EA-EC-11464-01	Second Level Undervoltage Time Delay Relays 162-153 and 62-154 Uncertainty Analysis	0
EA-EC-11464-02	First Level Undervoltage Relays 127-1 and 127-2 Drift Calculations	0
EA-EC-11464-03	First Level Inverse Time Undervoltage Relays 127-1 and 127-2 Uncertainty Analysis	0
EA-EC19401-01	Uncertainty Calculation for 2400V Safety Bus Voltage Meters	0
EA-EC3181717-01	Analysis of the Low Flow Testing of the Engineered Safeguards System Pumps	0

EA-ELEC08-0008	SIRW Tank Level Switch Position Uncertainty Calculation	0
EA-ELEC-AMP-016	Calculate the Cable Ampacity for Containment Spray	1
EA-ELEC-AMP-030	Capability of the 2400V Feeder Cables to Buses 1C and 1D from the Startup Transformer 1-2	2
EA-ELEC-CABLE-A1208-1	2.4KV Cable Ampacity for P-52B	0
EA-ELEC-CABLE-A1210-1	2.4KV Cable Ampacity for P-54A	0
EA-ELEC-EDSA-01	Auxiliary AC System EDSA Model Development and Verification and Validation	2
EA-ELEC-EDSA-03	LOCA with Offsite Power Available	1,2
EA-ELEC-EDSA-04	Second Level Undervoltage Relay Setpoint Determination (SLUR)	0
EA-ELEC-EDSA-05	Fast Transfer Analysis	2
EA-ELEC-EDSA-06	AC Short Circuit Analysis and EC 5000122058	2
EA-ELEC-LDTAB-005	Emergency Diesel Generators 1-1 and 1-2 Steady State Loadings	9,10
EA-ELEC-LDTAB-013	Worst Case Cable Loading Between SUT 1-2, and Buses 1C and 1D	0
EA-ELEC-LDTAB-019	Auxiliary Power System Measured Load Analysis	0
EA-ELEC-VOLT-010	Maximum and Minimum Station Power Voltages	0
EA-ELEC-VOLT-01A	Dynamic Response of Emergency Diesel Generators and ECCS Motor Acceleration Times	1
EA-ELEC-VOLT-006	Determine the Terminal Voltage at the 46 Safety-Related Motor Operated Valves	1
EA-ELEC-VOLT-033	Second Level Undervoltage Relay Setpoint	1
EA-ELEC-VOLT-034	Calculation of VT Burden and Ratio Correction Factor for 2400V Class 1E Buses	0
EA-ELEC-VOLT-036	Station Power Transformers No. 11, 12, 19, and 20 Tap Change -2.5 percent	0
EA-ELEC-VOLT-037	Degraded Voltage Calculation for the Safety-Related MOVs	3
EA-ELEC-VOLT-039	Stuck SGT 1-1 Tap Changer and Subsequent Station Power Voltages	0
EA-ELEC-VOLT-050	Motor Control Center Control Circuit Voltage Analysis	3
EA-ELEC-VOLT-051	MCC Power Circuit Minimum Required Voltage Analysis	1
EA-ELEC-VOLT-053	Preferred AC Voltage Analysis	0
EA-GL-8910-O1	Generic Letter 89-10 MOV Thrust Window Calculations	4

EA-GOTHIC-04-05	Development of LOCA Containment Response Base Deck	2
EA-GOTHIC-04-06	Development of MSLB Containment Response Base Deck	2
EA-GOTHIC-04-08	Containment Response to LOCA Using GOTHIC 7.2a	3
EA-GOTHIC-04-09	Containment Response to a MSLB Using GOTHIC 7.2a	3
EA-MOD-2005-004-03	ESS Flow Rates and Pump NPSH During Recirc Mode with CSS Throttling	4
EA-PIPEFLO-ESS-02	Pipe-Flo Professional 2007a Integrated Hydraulic Model of the Containment Spray, High Pressure and Low Pressure Safety Injection Systems	1
EA-PLTB-04	Pressure Looking and Thermal Binding Review for Appendix R Safe Shutdown Power Operated Gate Valves	0
EE-EDG-01	Evaluation of Maximum Diesel Room Temperature	0
MPR-4105	Technical Evaluation of EDG Frequency and Voltage Variations in Support of Operability Determination for Palisades	1
NB-PSA-ETSC	Mode 4 LOCA Load Recovery Period Basis.	4
NEI-1149-014	Palisades Design Basis AST MHA/LOCA Radiological Analysis	4
SUT1-2/SUT1-2/ALTC	Startup Transformer 1-2 Load Tap Changer Automatic Controls	0

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
CR-PLP-2014-04385	Merlin Drawing Revision Conflict For M0265BC Sheet 5	9/8/14
CR-PLP-2014-04406	Conditions Identified During A Tour Of The West Engineered Safeguards Room	9/10/14
CR-PLP-2014-04445	Indications Of White, Dry Boric Acid Inside The Insulation Box For LT-0332A, SIRW LEVEL TRANSMITTER, and LT-0332B, SIRW LEVEL TRANSMITTER	9/11/14
CR-PLP-2014-04446	NRC Inspector Identified a Degraded Jumper Connection On A Ground Cable	9/11/14
CR-PLP-2014-04450	Electrical Penetrations of Reactor Containment	9/11/14
CR-PLP-2014-04472	NRC Questions Whether it is Appropriate to Have a Design Basis Calculation Which States 3.6 Percent Air Entrainment Could Occur	9/12/14
CR-PLP-2014-04491	EA-ELEC-VOLT-39, Stuck SGT 1-1 TAP Changer and Subsequent Station Power Voltages, has not been updated	9/15/14
CR-PLP-2014-04495	NRC Inspectors Involved With the 2014 CDBI Questioned the Worst Case Load Assumption	9/15/14
CR-PLP-2014-04496	NRC Inspectors Involved With the 2014 CDBI Questioned the Worst Case Load Assumption	9/15/14

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
CR-PLP-2014-04413	Calculation EA-ELEC-VOLT-018, Which is Referenced In Design Basis Document 3.04, Could Not Be Found	9/10/14
CR-PLP-2014-04450	FSAR Unclear On Requirements for Electrical Protection for Containment Penetrations	9/11/14
CR-PLP-2014-04491	Calculation EA-ELEC-VOLT-039, Which Analyzes Maximum Voltage Change During Stuck Tap Condition for a Load Tap Changing Transformer, Does Not Bound Minimum Allowable Switchyard Voltage	9/15/14
CR-PLP-2014-04495	Non-Conservative Transient Load Analysis for 2.4kV Supply Breaker to Bus 1C	9/15/14
CR-PLP-2014-04496	Non-Conservative Transient Load Analysis for 2.4kV Supply Breaker to Bus 1D	9/15/14
CR-PLP-2014-04665	Process a Revision to Calculation EA-C-PAL-95-0877D to Address the Conditions Identified in Condition Reports CR-PLP-2014-04472 and CR-PLP-2014-04665, Includes Operability Evaluation of ESS Pumps	10/9/14
CR-PLP-2014-04672	As Presently Designed, The Thermal Overload Relays (TOR) For Six Safety-Related Motor Operated Valves Will Not Alarm Under Certain Overload Conditions	9/24/14
CR-PLP-2014-04679	The Current Range (325-375kw) Within RE-131 Doesn't Bound The Largest Calculation Load of P-8A	9/25/14
CR-PLP-2014-04680	The Current Range (325-375kw) Within RE-131 Doesn't Bound The Largest Calculation Load of P-66A	9/25/14
CR-PLP-2014-04681	Incorrect Acceptance Criteria Limits for Leak Testing of CV-3027 and CV-3056	9/25/14
CR-PLP-2014-04682	Time delay Between a Degraded Voltage Condition and Actuation of the Transfer from Offsite Power to the Diesel Generators is Ambiguous in Calculation EA-ELEC-EDSA-03	9/25/14
CR-PLP-2014-04696	Diesel Voltage and Frequency Limits Unanalyzed	9/26/14
CR-PLP-2014-04755	Improper Quality Classifications for Valve Sub-Components	9/30/14
CR-PLP-2014-04815	Segment of Non-Safety-Related Tubing Installed in Safety-Related System	10/2/14
CR-PLP-2014-04858	Lack of a Calculation that Establishes Total Time Delay, Including Loop Uncertainties, Between a Degraded Voltage Condition And Actuation of the Transfer from Offsite Power to the Diesel Generators	10/7/14
CR-PLP-2014-04860	The Cables from Start-Up Transformer 1-2 (EX-04) to Buses 1C (EA-11) and 1D (EA-12) are Undersized	10/7/14
CR-PLP-2014-04864	EC That Changed The 2400/480 Station Transformer Taps To A 2.5 Percent Boost Setting In 1995 Did Not Bound Worst Case High Voltage Conditions	10/8/14
CR-PLP-2014-04874	NRC Question on Flow Testing of Components in the Closed Cycle Cooling Water Program	10/8/14

CORRECTIVE ACTION DOCUMENTS GENERATED DUE TO THE INSPECTION

Number	Description or Title	Date
CR-PLP-2014-04881	Missed Surveillance For CS P-54A Due to Instruments Being Out Of Calibration Per The Requirements Of The ASME Code	10/8/14
CR-PLP-2014-04897	AOP-30, Loss of Shutdown Cooling, Attachment 8, Align Idle LPSI Pump Suction to the SIRW Tank When on Shutdown Cooling, May Not Be Adequate	10/9/14
CR-PLP-2014-04901	Transmission Network Model that Predicts the Change In Switchyard Voltage as a Result of a LOCA Does Not Consider Change in Plant Loading Imposed On Switchyard	10/9/14
CR-PLP-2014-04902	Motors That Are Undersized With Respect To Their Mechanical Loads Have Not Been Analyzed For Thermal Effects	10/9/14
CR-PLP-2014-04903	Degraded Voltage Monitor Total Delay-Time not in TS 3.3.5.2	10/09/14
CR-PLP-2014-04912	NRC Questions the Adequacy of the Testing Performed to Ensure E-60A/B Remain Qualified to the Capability Assumed During Design Basis Events	10/10/14
CR-PLP-2014-04915	Motor Operated Valve Calculation EA-GL-8910-01 Uses Obsolete Electrical Calculation EA-ELEC-VOLT-037 for Input Data Regarding Minimum Voltages to Valve Motors	10/10/14

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
A-NL-92-079	NRC Inspection Report 91-19 (EDSFI) Reply to Open Item 91-19-03, "Overloading of Buses 1C and 1D Feeder Cables from Startup Transformer"	3/29/95
A-NL-92-111	Spurious Diesel Generator Starts Due To Degraded Voltage Relay Time Delay Setting	3/9/95
A-PAL-92-066	Shutdown Cooling Heat Exchanger Performance Testing	12/16/92
AR 00870121	Inadequate Guidance for Operations When Outside Air Temperature Exceeds 95F	7/26/05
AR 199978	Revise PMS To Replace RO-98 Test Gauges With M&TE	4/28/14
CE 01066273-01	Excluded Required Equipment in OPR 108	12/7/06
C-PAL-94-0728JJ	No Heat Exchanger Performance Validation Testing is Performed on the Shutdown Cooling Heat Exchangers	12/9/96
C-PAL-94-0728W	Complete Assessment of Performance of Key Safety Systems Against the Plant Design Basis	5/2/95
C-PAL-97-1521	Heat Balance Discrepancies Recorded in Special Test T-365	12/8/97
CR-PLP-2006-05443	Level Switch LS-0329 Instrument Uncertainties	11/15/06
CR-PLP-2006-05479	EDG Room Cooling Intake Louvre Screens Dirty	11/16/06

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
CR-PLP-2006-05805	OPR 000108 Failed to Consider Elevated Temperatures on Several Components	12/7/06
CR-PLP-2006-05854	EDC Cooling Fan V-24C Discrepancies	12/11/06
CR-PLP-2006-05897	Design Calculation 01067491 Does Not Consider Frequency Variation in Calculating Load	12/31/06
CR-PLP-2007-02040	Level Switch LS-0329 Setpoint	5/15/07
CR-PLP-2008-04580	D/G Load Calc Did Not Account For Worst Case Load from CAC Fan Motors	11/7/08
CR-PLP-2008-04707	CDBI Team Questioned Using a Water Level Only 1 Inch Below Fuel Oil Suction Height	11/19/08
CR-PLP-2009-00477	Containment Spray Pump, P-54A, Has Very Little Clearance Between Shaft Coupling and Shaft Keyway	2/4/09
CR-PLP-2009-05533	CCW Pumps Lack Adequate Inservice Testing Margin	12/3/09
CR-PLP-2009-05534	Containment Spray Pumps (CSS) Lack Inservice Testing Margin	12/3/09
CR-PLP-2010-01331	NRC Resident Has Questions On QO-15	3/31/10
CR-PLP-2010-01734	Plant Response to IN 2010-09, Importance of Understanding Circuit Breaker Control Panel Indications	05/11/10
CR-PLP-2010-03083	Change in Gas Void Point/Size	7/27/10
CR-PLP-2010-05181	Technical Specification Test RO-98 Pump Test Results Low in Their Acceptance Band	10/17/10
CR-PLP-2010-06434	Transition of fire protection to NFPA 805	10/18/10
CR-PLP-2011-03087	Issue Identified During 2011 CDBI RFI 167	6/21/11
CR-PLP-2011-03221	Failure to Adequately Evaluate the Enclosure Installed Over the 1F/1G Buses	6/27/11
CR-PLP-2011-03419	Dripping Water in East Engineered Safeguards in the Vicinity of P-54A Containment Spray Pump	7/11/11
CR-PLP-2011-03835	Water Was Found Dripping from the Overhead East Engineered Safeguards Cooler in the Vicinity of P-54 A Spray Pump	8/4/11
CR-PLP-2011-03858	LOCA Could Occur in Mode 4	8/5/11
CR-PLP-2011-04749	Boric Acid Residue Discovered on Several Components	9/21/11
CR-PLP-2011-06313	No Uncertainty Analysis for Back-Leakage into SIRW Tank	11/17/11
CR-PLP-2012-02124	Extent of Condition for CR-PLP-11-6818, EQ Submergence Values Outside Containment	4/2/12

CORRECTIVE ACTION DOCUMENTS REVIEWED DURING THE INSPECTION

Number	Description or Title	Date
CR-PLP-2012-02382	Leakage from the Primary Coolant System to the SIRW Tank	4/10/12
CR-PLP-2012-05054	Potential Impact of Foreign Material Identified in CR PLP-2012-05049 on the ECCS Pumps	7/13/12
CR-PLP-2012-05553	Post-Accident Water Temperature in Containment Greater Than Air Temperature, Equipment EQ Needs Evaluation	8/8/12
CR-PLP-2012-06148	Lack of Complete Electrical Protective Coordination for Appendix R	9/11/12
CR-PLP-2012-06148	Electrical Coordination Will Not Support Transition to NFPA 805	9/11/12
CR-PLP-2012-06795	Shutdown Cooling Heat Exchanger (E-60A and E-60B) Inspections Removed from 1R23 Scope	10/18/12
CR-PLP-2013-00076	2014 CDBI Pre-Inspection Self-Assessment	12/23/13
CR-PLP-2013-01213	Diesel Generator Loading Calculation Non-Conservative	3/20/13
CR-PLP-2013-02674	RIS 2013-05; Relationship Between GDC and Tech Spec Operability	6/17/13
CR-PLP-2013-03894	AOV Calculations Not Revised	9/3/13
CR-PLP-2013-04003	Condensation from Room Cooler Dripping onto P-54A, CSS Pump and other NRC Identified Issues	9/10/13
CR-PLP-2013-04799	P-52B CCW Pump Has Oil Leak	11/6/13
CR-PLP-2013-04843	Green OOZE on Relay 150-1 Terminals	11/08/13
CR-PLP-2013-04940	As Found Thread Engagement Issues on Cooler for P-54A Containment Spray Pump	11/19/13
CR-PLP-2014-00042	Additional K-Line 480V Circuit Breakers Subject to ABB Part 21	1/03/14
CR-PLP-2014-00550	Air And Oil Leak on FCV-3057A	1/24/14
CR-PLP-2014-01679	P-54A Containment Spray Pump Low Differential Pressure During Performance of RO-98.	2/25/14
CR-PLP-2014-03841	No Evidence Found that Time Critical Operator Action Validation was Performed Since 2011.	7/23/14
CR-PLP-2014-04881	Missed Surveillance of CS Pump P-54A Risk-Evaluation	10/9/14

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VEN-M-107 Sheet 2278	Stress Isometric	10
VEN-M-107 Sheet 2281	Stress Isometric	10
VEN-M-107 Sheet 2282	Stress Isometric	4
VEN-M-241BC Sheet 33	Double Disc Gate Valve	0
VEN-M-241BC Sheet 35	Double Disc Gate Valve	0
VEN-M-255 Sheet 3	Drag Valve	0
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10 CFR 50.59 DOCUMENTS (SCREENINGS/SAFETY EVALUATIONS)

Number	Description or Title	Date
SDR-99-0884	Cancellation of TS Surveillance Procedures and Revision of FSAR	7/22/99

MISCELLANEOUS

Number	Description or Title	Date or Revision
	Letter from Dennis M. Crutchfield, NRC, to David P. Hoffman, Consumers Power Company, "SEP Topic VIII-4, Electrical Penetrations of Reactor Containment" (ADAMS Accession No. ML8104080152)	3/26/81
	Letter from Kerry A. Toner, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Electrical Penetrations of the Reactor Containment" (ADAMS Accession No. ML8210190459)	10/12/82
	Letter from Kerry A. Toner, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Status Update of Program to Evaluate the Adequacy of Penetration Protection from Overload and Short-Circuit Conditions" (ADAMS Accession No. ML8302240273)	2/11/83
	Letter from Robert A. Vincent, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Electrical Penetrations of Reactor Containment" (ADAMS Accession No. ML8106180170)	6/15/81
	Letter from Robert A. Vincent, Consumers Power Company, to Dennis M. Crutchfield, NRC, "SEP Topic VIII-4, Electrical Penetrations of the Reactor Containment" (ADAMS Accession No. ML8111200805)	11/16/81
	Letter from Thomas V. Wambach, NRC, to David J. VandeWalle, Consumers Power Company, "Integrated Plant Safety Assessment Report (IPSAR) Section 4.26, Electrical Penetrations of Reactor Containment – Palisades Plant" (ADAMS Accession No. ML8306160396)	6/10/83
	White Paper from Fauske and Associates: What Determines the Condition Of Significant Air Intrusion for the Draindown For Quiescent Water Inventories?	9/23/14
13-0327	Process Applicability Determination for EC 42422	0
1C-108-J9400-162-153	Protective Relay Setting – Bus 1C Second Level Under Voltage Time Delay Relay 162-153	4
1C-108-J9400-162-154	Protective Relay Setting – Bus 1D Second Level Under Voltage Time Delay Relay 162-154	3
70P-017	Engineering Specification for Shutdown Cooling Heat Exchanger	3
988055	P-54A Containment Spray Pump Oil Test Results	5/8/13

MISCELLANEOUS

Number	**Description or Title**	**Date or Revision**
988059	P-54A Containment Spray Pump Motor Inboard Bearing Oil Test Results	5/8/13
988060	P-54A Containment Spray Pump Motor Outboard Bearing Oil Test Results	7/1/13
ASCO Catalog 32	2, 3 and 4 Way Solenoid Valves Vendor Manual	0
ASME OMb Code-2003	Addenda to ASME OM Code-2001 Code for Operation and Maintenance of Nuclear Power Plants	8/29/03
Attachment 9.16	SNAPSHOT Assessment of Time Critical Operator Actions.	5/27/11
Attachment 9.4	SNAPSHOT Assessment of Time Critical Operator Actions.	9/3/14
CMT952010029	Enforcement Conference – Inoperable Diesel Generators – Inspection Report 94020	4/30/95
CR-PLP-2014-04881	Missed Surveillance of CS Pump P-54A Risk Evaluation	10/9/14
CV-3027	Diagnostic Testing Results – WO 52325715	4/24/12
CV-3031	IST – Valve Component Information	0
CV-3057	IST – Valve Component Information	0
DBD-1.01	Component Cooling Water System	8
DBD-1.01	Design Basis Document for Component Cooling Water System	8
DBD-2.01	Low Pressure Safety Injection System	10
DBD-2.01	Design Basis Document for Low Pressure Safety Injection System	10
DBD-2.02	Design Basis Document for High-Pressure Safety Injection System	9
DBD-2.03	Containment Spray System	8
DBD-2.03	Design Basis Document for Containment Spray System	8
DBD-3.04	Design Basis Document: 2400V AC System	7
DBD-3.05	Design Basis Document: 480V AC System	6
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DBD-4.04	Design Basis Document: Uninterruptible Power Supply	5
DBD-5.03	Design Basis Document: Emergency Diesel Generator Performance Criteria	8
DBD-5.04	Design Basis Document: Load Shedding Circuits	5
DBD-5.05	Design Basis Document: Design Basis Accident and Normal Shutdown Sequencer	7
DBD-6.01(TR)	Design Basis Document: Grid Interface Topical Report	4
DBD-6.02	Design Basis Document: 345KV Switchyard	4
DPR-20	Renewed Facility Operating License	251
DRN-08-01700	EOP Supplement 42, Not Tripping HPSI for Loss of Subcooling	9/5/08

MISCELLANEOUS

Number	Description or Title	Date or Revision
DRN-08-02217	Section 2.0 Post RAS Actions Step "N" Should be Identified as a Continuous Step (Circle C), so Operators can Continue to Perform Later Steps	12/2/08
DRN-12-01684	Commitments in Procedure do not Match Commitment List	5/14/12
DRN-14-01118	EOP Supplement 42	9/12/14
DRN-14-01222	SOP-3 Revision to Stage Equipment for Venting LPSI Discharge Piping with Valve MV-ES3420 per AOP-30, Attachment 8, Align Idle LPSI Pump Suction to the SIRW Tank When on SDC.	10/7/14
DWG. No. 5935 M-347 Sheet 14	Pressure Safety (Relief) Valve Data Sheet	2
EAR-99-0081	CVCS Declassification	0
LER 80-003	Licensee Event Report 80-003 – Containment Spray	5/13/80
Letter	Response to NUREG-0737	12/19/80
Letter (9002120116)	Response to Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment	1/29/90
Letter (ML052410206)	Supplementary Information for the Palisades Application for Renewed Operating License Resulting from Aging Management Programs Audit	8/25/05
LR-AMPBD-06-CCCW	Closed Cycle Cooling Water Program	3
LTR-PSA-14-01	Mode 4 LOCA Analysis	10/7/14
M0001GD 0999	Fabricating Engineers Inc. Technical Manual for Shutdown Cooling Heat Exchanger	0
ML050940446	Palisades Nuclear Plant Application for Renewed Operating License	3/22/05
ML071800216	License Amendment Request: Replacement of Containment Sump Buffer	6/28/07
NB-PSA-SY-CSS	Palisades Nuclear Plant Probabilistic Safety Assessment System Notebook – Containment Spray System	0
PL-CCW	Component Cooling Water	7
PL-CSS	Containment Spray System Lesson Plan – CV3001, CV3002, and CV-3030	9
PL-CSS	Containment Spray System	9
PLJPM-LOR-SI-14	JPM - PERFORM PRE-RAS ACTIONS IAW EOP SUPPLEMENT 42	0
PLJPM-LOR-SI-15	JPM - PERFORM PRE-RAS ACTIONS IAW EOP SUPPLEMENT 42	0
PLJPM-LOR-SI-16	JPM - PERFORM POST RAS ACTIONS	2
PLJPM-LOR-SI-17	JPM - PERFORM POST RAS ACTIONS	1

MISCELLANEOUS

Number	Description or Title	Date or Revision
PLJPM-LOR-SI-18	JPM - PERFORM POST RAS ACTIONS	2
PLJPM-LOR-SI-19	JPM - PERFORM POST RAS ACTIONS	1
PLSEG-LOR-13F-02	Scenario – AOP 30	4
PLSEG-LOR-13G-02	Scenario – EOP 4.0	0
PLSEG-LOR-14-CDBI-1	EOP-4.0 LBLOCA Scenario for 2014 CDBI.	9/25/14
PLSEG-LOR-14-CDBI-2	Mode 4 LOCA Scenario for 2014 CDBI.	10/7/14
PLSEG-LOR-LOCA	Scenario – EOP 4.0 Loss of Coolant Accident	0
PL-SIS	Safety Injection System	6
QO-16	Basis Document for QO-16: Inservice Test Procedure – Containment Spray Pumps	16
Reply EC 53093	Evaluation For Potential Air Entrainment at the Lowest Levels of SIRW Tank Inventory Considering A Single Failure More limiting than the One Considered in EA-C-PAL-95-087DD	9/24/14
RI-38 Basis	Basis Document for SIRW Tank Level Instrument Calibration	10
SDR-99-0884	Cancellation of TS Surveillance Procedures and Revision of FSAR	7/22/99
SEP-APJ-PLP-101	Mechanical Containment Penetration Basis	0
SEP-ISI-PLP-002	ASME Code Boundaries for ASME Section XI Inservice Inspection Program	1
SEP-ISI-PLP-002	ASME Code Boundaries For Section XI Inservice Inspection Program	1
SEP-PLP-IST-102	Inservice Testing of Selected Safety-Related Pumps	0
SOP-15	Service Water System	0
USI A-46	Equipment Evaluation Report	5/3/95

MODIFICATIONS

Number	Description or Title	Date or Revision
FC-718	RAS Logic Seal-In Circuit Addition	12/19/86

OPERABILITY EVALUATIONS

Number	Description or Title	Date or Revision
CR-PLP-2014-4696 CA-1	Operability for all SSCs Within Maximum Variations of EDG Frequency and Voltage Allowed by Technical Specifications	10/22/14
OPR 000108	Operability Recommendation – EDG Room Temperature	2

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Administrative Procedure 4.06	Emergency Operating Procedure Development and Implementation	21
Administrative Procedure 4.66	Abnormal Operating Procedure Development and Implementation	5
Administrative Procedure TCA	Control of Time Critical Operator Actions	3
AOP-23	Primary Coolant Leak	1
AOP-30	Loss of Shutdown Cooling	1
AOP-36	Loss Of Component Cooling	0
AOP-37	Loss of Instrument Air	0
ARP-4	Reactor Water Level Low Alarm Response Procedure	62
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EN-DC-195	Margin Management	7
EN-DC-219	Gas Accumulation Management	3
EN-MA-145	Maintenance Standard for Torque Applications	3
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EOP 4.0	Loss of Coolant Accident Recovery	23
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EOP Supplement 42	Pre and Post RAS Actions	7
EOP TCA	EOP Time Critical Operator Action Basis	0
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ESSO-1	Containment Spray Header Fill	16

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MO-7A-1	Emergency Diesel Generator 1-1	86
MO-7A-2	Emergency Diesel Generator 1-2	86
NB-PSA-SM	Probabilistic Safety Assessment Model Summary System Notebook	2
NB-PSA-SSS	SIRW Tank and Containment Sump Suction System Probabilistic Safety Assessment System Notebook	1
NPOA	Nuclear Plant Operating Agreement	1
ONP-17	Loss of Shutdown Cooling (SDC)	40
QO-16	Inservice Test Procedure – Containment Spray Pumps	35
QO-2	Recirculation Actuation System	44
QO-5	Valve Test Procedure (Includes Containment Isolation Valves)	89
RE-131	Diesel Generator 1-1 Load Reject	6
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Number	**Description or Title**	**Revision**
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SURVEILLANCES (COMPLETED)

Number	**Description or Title**	**Date or Revision**
T-205-A	East Engineering Safeguards High Pressure Air System Performance Verification	11/23/14
T-205-B	West Engineering Safeguards High Pressure Air System Performance Verification	12/2/14
T-278-3A	Nitrogen Station 3A Performance Test	2/19/14
WO-PLP-52435851	LPSI and Containment Spray Comprehensive Pump Test and Check Valves Test	2/25/14
WO-PLP-52547674	Inservice Test Procedure – Containment Spray Pumps	5/29/14

WORK ORDERS

Number	**Description or Title**	**Date or Revision**
178171-01	Clean EDG Room Cooling Air Inlet Louvre Screens	11/21/06
342334-01	Change SUT 1-2 (EX-04) Tap Changer Setpoint	
51521714-01	Clean Door Louvre Screens	11/21/06
51521714-02	Repair Door Louvre Screens	11/21/06
51521714-03	Clean Wall Louvre Screens	11/21/06
52325512-01	RI-14, SIRW Tank Level Switch Interlock Test	8/6/12
52430162-01	RI-14, SIRW Tank Level Switch Interlock Test	3/6/14
52432473-01	RT-92 – Inspect ECCS Train Cont Sump Suction Inlet	3/9/14
52435963-01	Containment Sump Clean-Out and Inspection	2/26/14
WT-PLP-2010-00261	Update DBD 3.04 Table 3-2: "2400V AC Supply Cables, Design Ratings, Loading and Margins	6/2/10
WT-PLP-2014-00216	WT Written to Document Actions Resulting from CDBI Focused Area Self-Assessment LO-PLP-2013-00076	7/17/14

LIST OF ACRONYMS USED

°C	Celsius Degrees
°F	Fahrenheit Degrees
AC	Alternating Current
ADAMS	Agencywide Document Access Management System
AOP	Abnormal Operating Procedure
AOV	Air-Operated Valve
AR	Action Request
ARM	Area Radiation Monitor
ASME	American Society of Mechanical Engineers
BHP	Brake Horsepower
CAP	Corrective Action Program
CCW	Component Cooling Water
CDBI	Component Design Bases Inspection
CFR	Code of Federal Regulations
CS	Containment Spray
CVCS	Chemical and Volume Control
DBD	Design Basis Document
DC	Direct Current
DRS	Division of Reactor Safety
EC	Engineering Change
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
EPU	Extended Power Uprate
EQ	Equipment Qualifications
ESS	Engineered Safeguards System
FIN	Finding
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GL	Generic Letter
gpm	Gallons per Minute
HELB	High Energy Line Break
HPSI	High Pressure Safety Injection
HVAC	Heating, Ventilation and Air-Conditioning
IEEE	Institute of Electrical and Electronic Engineers
IMC	Inspection Manual Chapter
IN	Information Notice
IP	Inspection Procedure
IR	Inspection Report
IR	Issue Report
ISI	Inservice Inspection
IST	Inservice Testing
kV	Kilovolt
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MCC	Motor Control Center
MOV	Motor-Operated Valve

NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
PARS	Publicly Available Records System
PCS	Primary Coolant System
P&ID	Piping and Instrumentation Drawing
PI&R	Problem Identification and Resolution
PRA	Probabilistic Risk Assessment
psi	Pounds Per Square Inch
psid	Pounds Per Square Inch Differential
psig	Pounds Per Square Inch Gauge
RIS	Regulatory Issue Summary
SDC	Shutdown Cooling
SDP	Significance Determination Process
SEP	Systematic Evaluation Program
SG	Steam Generator
SIRW	Safety Injection Refueling Water
SR	Surveillance Requirement
SSC	Systems, Structures, and Components
TI	Temporary Instructions
TS	Technical Specification
URI	Unresolved Item
Vac	Volts Alternating Current
VAC	Volts Alternating Current
VDC	Volts Direct Current
Vdc	Volts Direct Current
WO	Work Order

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Sincerely,

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Engineering Branch 2
Division of Reactor Safety

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