UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE RD. SUITE 210
LISLE, IL 60532-4352

February 12, 2016

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 – NRC INTEGRATED INSPECTION REPORT 05000255/2015004

Dear Mr. Vitale:

On December 31, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palisades Nuclear Plant. The enclosed report documents the results of this inspection, which were discussed on January 12, 2016, with yourself, and other members of your staff.

Based on the results of this inspection, the NRC has identified four issues that were evaluated under the risk significance determination process as having a very-low safety significance (Green). The NRC has also determined that violations are associated with these issues. These violations are being treated as Non-Cited Violations (NCVs), consistent with Section 2.3.2 of the Enforcement Policy. These NCVs are described in the subject inspection report. Additionally, a 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 copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at 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 Palisades Nuclear Plant.

In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of the NRC's Agencywide Documents Access and Management System (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

John Jandovitz, Acting Chief
Branch 3
Division of Reactor Projects

/RA/

Christine A. Lipa
Deputy Division Director,
Division of Nuclear Materials Safety

Docket No. 50–255
License No. DPR–20

Enclosure:
IR 05000255/2015004

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–255
License No:	DPR–20
Report No:	05000255/2015004
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	October 1 through December 31, 2015
Inspectors:	A. Nguyen, Senior Resident Inspector J. Boettcher, Resident Inspector A. Armstrong, Reactor Operations Engineer J. Cassidy, Senior Health Physicist J. Corujo-Sandin, Engineering Inspector, Mechanical M. Domke, Observer G. Hansen, Senior Emergency Preparedness Inspector M. Holmberg, Reactor Inspector C. Hunt, Reactor Engineer M. Keefe-Forsythe, Human Factors Specialist J. Lennartz, Project Engineer V. Myers, Senior Health Physicist G. O'Dwyer, Reactor Engineer L. Rodriguez, Engineering Inspector, Mechanical J. Rutkowski, Project Engineer T. Taylor, D.C. Cook Resident Inspector
Approved by:	J. Jandovitz, Acting Chief Branch 3 Division of Reactor Projects Christine A. Lipa Deputy Division Director, Division of Nuclear Materials Safety

Enclosure

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SUMMARY

Inspection Report (IR) 05000255/2015004, October 1, 2015 – December 31, 2015; Palisades Nuclear Plant; Inservice Inspection Activities; Component Design Bases Inspection.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Two Green findings and two Severity Level IV traditional enforcement violations were identified by the inspectors. The findings were considered non-cited violations (NCVs) of U.S. Nuclear Regulatory Commission (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 Inspection Manual Chapter (IMC) 0609, "Significance Determination Process (SDP)," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. 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.

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of Title 10, *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion IX, "Control of Special Processes," for the licensee's failure to perform a dye penetrant (PT) examination of the Safety Injection System (SIS) pipe lug welds in accordance with the American Society of Mechanical Engineers (ASME) Code Section XI requirements. The licensee entered this issue into the Corrective Action Program (CAP) as CR-PLP-2015-04191, repeated the PT examination of the affected SIS lug welds to meet the full extent of coverage required by the ASME Code, repeated examinations of other welds conducted by the PT examiner during the outage, and removed the PT examiner from further weld examination activities.

  This performance deficiency was determined to be more than minor because, if left uncorrected, the failure to perform a PT examination in accordance with the ASME Code requirements could result in acceptance and return to service of a component with an undetected crack that would increase the possibility of pipe leakage or failure. In addition, the failure to perform a PT examination in accordance with the ASME Code adversely affected the Mitigating System Cornerstone attribute of Equipment Performance, because it could result in failure to detect cracks in pipe welds, which would reduce the availability and reliability of the SIS mitigating system. The inspectors evaluated the finding in accordance with IMC 0609, Appendix A, "The SDP for Findings At-Power," Exhibit 2, "Mitigating Systems Screening Questions," and answered "yes" to screening question number 1. Although this finding adversely affected the design or qualification of the SIS pipe lugs, the finding screened as very-low safety significance (Green), because it did not result in the loss of operability or functionality of the affected SIS pipe segment. This finding had a cross-cutting aspect in the Field Presence component of the Human Performance cross-cutting area. Specifically, licensee leaders were not observed in the work areas of the plant to coach and reinforce standards or expectations for the licensee's vendor staff to ensure deviation from standards and expectations were promptly corrected [H.2]. (Section 1R08.1)

- Green. The inspectors identified a finding of very-low safety significance, and an associated NCV of 10 CFR, Part 50, Appendix B, Criterion II, "Quality Assurance Program," for the licensee's failure to identify all component cooling water (CCW) structures, systems, and components (SSC), which were required to be covered by the Quality Assurance Program (i.e., be safety-related). As a result, the licensee incorrectly credited nonsafety-related CCW components to remain functional during and following a design basis event (DBE). The licensee entered this finding into their CAP and, after performing operability determinations, concluded the system would still be capable of performing its function.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance, and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding screened as having very-low safety significance (Green) because, although it was a deficiency affecting the design or qualification of a mitigating SSC, the SSC maintained its operability. The inspectors did not identify a cross-cutting aspect associated with this finding because it was determined not to be representative of current performance. (Section 1R21.b.(1))

- SL IV. The inspectors identified a Severity Level (SL) IV, NCV of 10 CFR, Part 50, Section 59, "Changes, Tests, and Experiments," for the licensee's failure to maintain records of written safety evaluations, which provide the bases for concluding the nonsafety-related portions of the CCW system inside containment could be credited to perform their function during and following a DBE, and that the change would not result in an unreviewed safety question. The licensee entered this issue into their CAP and, after performing operability determinations, concluded the system would still be capable of performing its function.

  The violation was determined to be more than minor because the inspectors could not reasonably determine that the changes would not have ultimately required NRC prior approval. The violation was categorized as a SL IV in accordance with Section 6.1.d.2 of the NRC Enforcement Policy because the resulting changes were evaluated by the SDP as having very-low safety significance (i.e., green finding). The resulting changes, the violation's underlying technical concerns, impacted the Mitigating Systems cornerstone, and were evaluated separately as the Green finding with the associated 10 CFR, Part 50, Appendix B, Criterion II, NCV discussed above. The inspectors did not identify a cross-cutting aspect because cross-cutting aspects are not assigned to traditional enforcement violations. (Section 1R21.b.(2))

- SL IV. The inspectors identified a SL IV, NCV of 10 CFR, Part 50.59, "Changes, Tests, and Experiments," and an associated finding of very-low safety significance (Green) for the licensee's failure to maintain a record of the declassification of the Chemical Volume and Control System (CVCS) from safety-related to nonsafety-related, which includes a written evaluation that provides the bases for the determination that the change did not require a license amendment. The licensee entered this issue into their CAP, and after a review of the system, determined there was reasonable assurance that it could perform its function.

The inspectors determined the underlying technical concern was a performance deficiency associated with the Mitigating Systems cornerstone that was more than minor because, if left uncorrected, would become a more significant safety concern. The underlying technical concern screened as a finding with very-low safety significance (Green) because, although it affected the design or qualification of the CVCS, it did not result in the loss of functionality of the CVCS. The violation was determined to be more than minor because the inspectors could not reasonably determine that the changes would not have ultimately required NRC prior approval. The violation was categorized as a SL IV in accordance with Section 6.1.d.2 of the NRC Enforcement Policy because the changes were evaluated by the SDP, described above, as having very-low safety significance (i.e., Green finding). The inspectors did not identify a cross-cutting aspect associated with the finding because the finding was not representative of current performance. (Section 1R21.b.(3))

- 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 CAP. These violations and CAP tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The plant began the assessment period shut down for a planned refueling outage (RFO) 1R24. On October 18, 2015, the reactor was taken critical and the plant was synchronized to the grid on October 19, 2015. The reactor achieved full power on October 22, 2015, and remained at or near full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition - High Wind Conditions

a. Inspection Scope

On November 12 and 13, 2015, thunderstorms with potential high winds were forecast in the vicinity of the facility. The inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. On November 11 and 12, 2015, the inspectors walked down the Emergency Diesel Generators (DGs), Service Water System (SWS), and Auxiliary Feedwater (AFW) System, in addition to the licensee's emergency alternating current (AC) power systems, because their safety-related functions could be affected or required as a result of high winds or tornado-generated missiles or the loss of offsite power. The inspectors evaluated the licensee staff's preparations against the site's procedures and determined that the staff's actions were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles during a tornado. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. The inspectors also reviewed a sample of Corrective Action Program (CAP) items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the CAP in accordance with station corrective action procedures. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in Inspection Procedure (IP) 71111.01–05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

The inspectors performed partial system walkdowns of the following risk-significant systems:

- Spent Fuel Pool (SFP) Cooling and Ventilation;
- Boric Acid addition pathway; and
- Electrical Power system alignment during 1R24.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

These activities constituted three partial system walkdown samples as defined in IP 71111.04–05.

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Area 24: AFW pumps room;
- Fire Area 13G: SFP heat exchanger room;
- Fire Area 17: Refueling and SFP area;
- Fire Area 13B: Charging pump rooms; and
- Fire risk-significant areas for the higher risk plant operating state #2 during 1R24.

The inspectors reviewed areas to assess if the licensee had implemented a Fire Protection Program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan. The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

These activities constituted five quarterly fire protection inspection samples as defined in IP 71111.05–05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's testing of VHX-1, #1 Containment Air Cooler, to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. The inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this document.

This annual heat sink performance inspection constituted one sample as defined in IP 71111.07-05.

b. Findings

No findings were identified.

.2 Triennial Review of Heat Sink Performance (71111.07T)

a. Inspection Scope

The inspectors reviewed operability determinations, completed surveillances, vendor manual information, associated calculations, performance test results, and cooler inspection results associated with the 1-2 DG lube oil cooler (E-31B). This cooler was chosen based on its risk significance in the licensee's probabilistic safety analysis, its important safety-related mitigating system support function, its operating history, and its relatively low margin.

The licensee did not do thermal performance testing for E-31B. However, the inspectors verified that inspection, maintenance, and monitoring of biotic fouling and macrofouling programs were adequate to ensure proper heat transfer. This was accomplished by verifying: (1) the methods used were consistent with accepted industry practices, or an equivalent; (2) the cleanings were consistent with the selected methodology; (3) the inspection acceptance criteria were consistent with procedure requirements; and (4) results of inspections were adequate.

For E-31B, the inspectors reviewed the methods and results of heat exchanger performance inspections. The inspectors verified the methods used to inspect and clean the heat exchanger were consistent with as-found conditions identified, expected degradation trends, and industry standards and contained established acceptance criteria. The as-found results were recorded, evaluated, and appropriately dispositioned such that the as-left condition was acceptable.

In addition, the inspectors verified the condition and operations of E-31B were consistent with design assumptions in heat transfer calculations and as described in the UFSAR. This included verification that the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors verified the licensee evaluated the potential for water hammer and established adequate controls and operational limits to prevent heat exchanger degradation due to excessive flow-induced vibration during operation. In addition, Eddy Current test reports and visual inspection records were reviewed to determine the structural integrity of the heat exchanger.

The inspectors also reviewed the licensee's operation of the SWS and Ultimate Heat Sink. This included a review of the licensee's procedures for a loss of service water and verification that instrumentation, which is relied upon for decision-making, was available and functional. In addition, the inspectors verified that macrofouling was adequately monitored, trended, and controlled by the licensee to prevent clogging. The inspectors verified that the licensee's biocide treatments for biotic control were adequately conducted and the results were monitored, trended, and evaluated. The inspectors also reviewed strong pump-weak pump interaction and design changes to the SWS and Ultimate Heat Sink.

The inspectors performed a system walkdown of the SWS intake structure to verify the licensee's assessment on structural integrity and component functionality. This included verification that the licensee ensured proper functioning of traveling screens and strainers and structural integrity of component mounts. In addition, the inspectors verified that the Service Water pump bay silt accumulation was monitored, trended, and maintained at an acceptable level by the licensee, and that water level instruments were functional and routinely monitored. The inspectors also verified the licensee's ability to ensure functionality of the bay and instruments during adverse weather conditions and that the licensee had adequate protection against silt introduction during periods of low-flow or low water level.

Finally, the inspectors reviewed condition reports related to heat exchangers/coolers and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of corrective actions. The documents that were reviewed are included in the Attachment to this report.

These inspection activities constituted two triennial heat sink inspection samples as defined in IP 71111.07-05.

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08P)

From September 21, 2015, through October 7, 2015, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, steam generator (SG) tubes, emergency feedwater systems, risk-significant piping and components and containment systems.

The inspections described in Sections 1R08.1, 1R08.2, R08.3, IR08.4, and 1R08.5 below constituted one ISI sample as defined in IP 71111.08.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors either observed or reviewed data acquired for the following non-destructive examinations (NDEs) mandated by Title 10, *Code of Federal Regulations* (CFR), Part 50.55a, "Codes and Standards," to evaluate compliance with the applicable American Society of Mechanical Engineers (ASME) Code requirements, and if any indications and defects were detected, to determine whether these were dispositioned in accordance with the ASME Code or an U.S. Nuclear Regulatory Commission (NRC)-approved alternative requirement.

- Automated Phased Array Ultrasonic (UT) examination of Primary Coolant System (PCS) Hot Leg Drain Nozzle Weld (PCS-42-RCL-1H-3/2);
- Automated Phased Array UT examination of PCS Cold Leg Drain Nozzle Weld (PCS-30-RCL-2A-5/2);
- Automated Phased Array UT examination of PCS Cold Leg Charging Nozzle Weld (PCS-30-RCL-1A-11/2);
- Automated Phased Array UT examination of Shutdown Cooling Hot Leg B Nozzle to Pipe Weld (PCS-12-SDC-2H1-2);
- Dye Penetrant (PT) examination of Safety Injection System (SIS) Pipe Lug Welds (ESS-12-SIS-1A1-3PL1-4); and
- Magnetic Particle examination of 'A' SG, E-50A, Support Skirt Weld (1-110-251).

For the surface and volumetric NDE performed since the previous RFO, the licensee had not identified any relevant indications. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed the following pressure boundary welds completed for risk-significant systems since the beginning of the last RFO to determine if the licensee applied the pre-service NDEs and acceptance criteria required by the Construction Code and ASME Code, Section XI. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine whether the weld procedures were qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Weld repair/replacement of Class 1 Pressurizer Spray connection elbow (Welds PCS-3-PSS-2A1-1X1 and 3X1)

b. Findings

(1) Inadequate Dye Penetrant Examination of Pipe Lug Welds

Introduction: The inspectors identified a finding of very-low safety significance (Green), and an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," for the licensee's failure to perform a PT Examination of SIS pipe lug welds in accordance with the ASME Code Section XI requirements.

Description: During observation of a PT examination of the SIS pipe lug welds, the inspectors identified that the extent of the licensee's examination coverage was less than that required by the ASME Code. The inspectors were concerned that absent NRC intervention, the weld examinations, as performed, would be inadequate to detect service-induced cracks.

The inspectors observed the licensee's vendor examiner during PT examination of SIS pipe lug welds ESS-12-SIS-1A1-3PL-1, 2, 3 and 4 in accordance with site procedure CEP-NDE-0641, "Liquid Penetrant Examination for ASME Section XI." Procedure CEP-NDE-0641 was identified as "Informational Use," and, as such, was not required to be present at the examination location. Additionally, this procedure referenced, but did not include, the ASME Code Section XI figure numbers which defined the extent of examination coverage for these welds. The licensee's examiner elected to bring a copy of the procedure to the weld location, but not a copy of the ASME Code figure that defined the extent of the examination. Following application of the penetrant material on the pipe lug welds, the inspectors identified areas of base metal adjacent to these lug welds without penetrant material coverage. After prompting by the NRC, the licensee's examiner added penetrant material to obtain coverage for 0.25 inches of adjacent base material. Similarly, after application of developer, the inspectors noted areas which did not receive developer coverage in the examination area. The licensee's examiner corrected this error by application of developer to obtain coverage that included 0.25 inches of the adjacent base material. Following application of the developer, the inspectors identified penetrant material "bleeding out" from creviced locations near the ends of the lug welds caused by the weld configuration/geometry. The inspectors were concerned that these "bleed out" locations could have masked/obscured crack indications and result in a PT examination with less than the full extent of required coverage. The licensee's examiner initially reported that these areas did represent limitations, but this was an expected outcome for these welds. The inspectors' concern for masked areas prompted the licensee's examiner to obtain measuring equipment and record the extent of the examination area limited by the excessive "bleed out" near the end of these lug welds. Following completion of the PT examination for these lug welds, the inspectors identified that the extent of examination coverage, as completed, included only 0.25 inches of base material adjacent to the welded lugs, as mentioned above, which was not sufficient to meet the ASME Code Section XI. Specifically, for these lug welds, the extent of examination as defined by the ASME Code Section XI, Figure IWC-2500-5, included 0.5 inches of the base metal adjacent to each side of the weld.

The licensee entered this issue into the CAP as CR-PLP-2015-04191, repeated the PT examination of the affected SIS lug welds to meet the full extent of coverage required by the ASME Code, repeated examinations of other welds conducted by the PT examiner during the outage, and removed the examiner of these pipe lug welds from further weld examinations. Because the NRC identified this issue and the licensee corrected this issue before the SIS lug welds were returned to service, the operability of the SIS was not affected.

Analysis: The inspectors determined that the licensee's failure to perform a PT examination of the SIS lug welds in accordance with the ASME Code Section XI requirements was contrary to 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," and was a performance deficiency. The inspectors determined that the performance deficiency was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, Appendix B, "Issue Screening," dated September 7, 2012. Specifically, if left uncorrected, the failure to perform a PT examination in accordance with the ASME Code requirements could result in acceptance and return to service of a component with an undetected crack that would increase the possibility of pipe leakage or failure. In addition, the failure to perform a PT examination in accordance with the ASME Code adversely affected the Mitigating System cornerstone attribute of Equipment Performance because it could result in failure to detect cracks in pipe welds, which would reduce the availability and reliability of the SIS mitigating system.

The inspectors evaluated the finding in accordance with IMC 0609, "Significance Determination Process (SDP)," Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings," and Exhibit 2, "Mitigating Systems Screening Questions" of IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," dated July 1, 2012. The inspectors answered "yes" to screening question number 1 of exhibit 2. Although this finding adversely affected the design or qualification of the SIS pipe lugs, the finding screened as very-low safety significance (Green), because it did not result in the loss of operability or functionality of the affected SIS pipe segment.

This finding had a cross-cutting aspect in the Field Presence component of the Human Performance cross-cutting area. Specifically, licensee leaders were not observed in the work areas of the plant to coach and reinforce standards or expectations for the licensee's vendor staff to ensure deviation from standards and expectations were promptly corrected. [H.2]

Enforcement: Title 10 CFR, Part 50, Appendix B, Criterion IX, "Control of Special Processes," requires that measures shall be established to assure that special processes, including welding, heat treating, and non-destructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. The ASME Code Section XI, Article IWC-2500, "Examination and Pressure Test Requirements," states "components shall be examined and pressure tested as specified in Table IWC-2500-1." Table IWC-2500-1, "Examination Categories," Category C-C, "Welded Attachments for Vessels, Piping, Pumps, and Valves," Item C3.20, "Welded Attachments – Piping," requires a 100 percent surface examination of each welded attachment in accordance with Figure IWC-2500-5. Figure IWC-2500-5 depicts the attachment weld and 0.5 inches of base metal adjacent to each side of the weld.

Contrary to the above, on September 23, 2015, during PT examination of pipe lug welded attachments classified as Item C3.20 (welded pipe lugs ESS-12-SIS-1A1-3PL-1, 2, 3 and 4), the licensee completed examination of approximately 0.25 inches of the base metal adjacent to the attachment welds. Once identified, the licensee repeated the PT examination to include the full extent of the required area; 0.5 inches of base metal. Because this violation was of very-low safety significance, and was entered into the licensee's CAP as CR-PLP-2015-04191, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy.
(NCV 05000255/2015004-01; Inadequate Dye Penetrant Examination of Pipe Lug Welds).

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

A bare metal visual (BMV) examination and a non-visual examination were required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors reviewed the records of the BMV examination of the reactor vessel head at each of the penetration nozzles to determine whether the activities were conducted in accordance with the requirements of ASME Code Case (CC) N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, to determine:

- if the required visual examination scope/coverage was achieved and limitations (if applicable were recorded), in accordance with the licensee procedures;
- if the licensee criteria for visual examination quality and instructions for resolving interference and masking issues were adequate; and
- for indications of potential through-wall leakage, that the licensee entered the condition into the corrective action system and implemented appropriate corrective actions.

The inspectors observed a number of non-visual examinations conducted on the reactor vessel head penetrations to determine whether the activities were conducted in accordance with the requirements of ASME CC N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, to determine:

- if the required examination scope (volumetric and surface coverage) was achieved and limitations (if applicable were recorded), in accordance with the licensee procedures;

- if the UT examination equipment and procedures used were demonstrated by blind demonstration testing;
- for indications or defects identified, that the licensee documented the conditions in examination reports and/or entered this condition into the corrective action system and implemented appropriate corrective actions; and
- for indications accepted for continued service, that the licensee evaluation and acceptance criteria were in accordance with the ASME Section XI Code, 10 CFR 50.55a(g)(6)(ii)(D) or an NRC-approved alternative.

The licensee did not perform any welded repairs to vessel head penetrations since the beginning of the preceding outage. Therefore, no NRC review was completed for this inspection procedure attribute.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

The inspectors performed a walk down of the PCS and related lines in the containment during the licensee's boric acid walk down to verify whether the licensee's boric acid corrosion control visual examinations emphasized locations where boric acid leaks can cause degradation of safety significant components.

The inspectors reviewed the following licensee evaluations of PCS components with boric acid deposits to determine if degraded components were documented in the CAP. The inspectors also evaluated corrective actions for any degraded PCS components to determine if they met the ASME Section XI Code.

- 14-PAL-0032; P-50B, Primary Coolant Pump (PCP) seal leak;
- 14-PAL-0187; CV-1057, Pressurizer Spray Valve packing leak; and
- 14-PAL-0099; MV-PC600, 'B' SG Hot Leg sample line isolation valve, downstream cap leak.

The inspectors reviewed the following corrective actions related to evidence of boric acid leakage to determine if the corrective actions completed were consistent with the requirements of the ASME Code Section XI and 10 CFR Part 50, Appendix B, Criterion XVI.

- CR-PLP-2014-0343; Boric Acid on P-50A, 'A' PCP; and
- CR-PLP-2014-02127; Boric Acid on MV-PC1038A, Pressurizer pressure transmitter/level transmitter root valve, fitting.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed acquisition and analysis of Eddy Current testing (ET) data, interviewed ET data analysts, and reviewed documentation related to the SG ISI Program to determine if:

- in-situ SG tube pressure testing screening criteria used were consistent with those identified in the Electric Power Research Institute (EPRI) TR-1025132, "SG In-Situ Pressure Test Guidelines," and that these criteria were properly applied to screen degraded SG tubes for in-situ pressure testing;
- the numbers and sizes of SG tube flaws/degradation identified was bounded by the licensee's previous outage operational assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to meet the TSs, and the EPRI 1003138, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 6;
- the SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;
- the licensee identified new tube degradation mechanisms and implemented adequate extent of condition inspection scope and repairs for the new tube degradation mechanism;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;
- the licensee implemented an inappropriate "plug on detection" tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gallons per day or the detection threshold during the previous operating cycle;
- the ET probes and equipment configurations used to acquire data from the SG tubes were qualified to detect the known/expected types of SG tube degradation in accordance with Appendix H, "Performance Demonstration for ET Examination," of EPRI 1013706, "Pressurized Water Reactor Steam Generator Examination Guidelines," Revision 7; and
- the licensee performed secondary side SG inspections for location and removal of foreign materials.

The licensee did not perform in-situ pressure testing of SG tubes. Therefore, no NRC review was completed for this inspection attribute.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine whether:

- the licensee had established an appropriate threshold for identifying ISI-related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review of Licensed Operator Requalification (71111.11Q)

a. Inspection Scope

On December 1, 2015, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training. The inspectors verified that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and that training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly Licensed Operator Requalification Program simulator sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.2a Resident Inspector Quarterly Observation During Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On October 7, 2015, the inspectors observed the control room operating crew conduct a drain of the PCS to reduced inventory to facilitate work during RFO 24. This was an activity that required heightened awareness and precise plant control and was an increased risk period (Yellow) for plant operations. The inspectors evaluated the following areas:

- licensed operator performance;
- the crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations; and
- oversight and direction from supervisors.

Performance in these areas was compared to pre-established operator action expectations, procedural compliance, and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

.2b Resident Inspector Quarterly Observation During Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

On October 18, 2015, the inspectors observed a reactor startup and approach to criticality following RFO 24. This was an activity that required heightened awareness, precise plant control, and was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- the crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations; and
- oversight and direction from supervisors.

Performance in these areas was compared to pre-established operator action expectations, procedural compliance, and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

a. Findings

No findings were identified.

.2c Resident Inspector Quarterly Observation During Periods of Heightened Activity or Risk (71111.11Q)

a. Inspection Scope

October 16, 2015, the inspectors observed the control room operating crew respond to a voltage transient which caused a loss of incoming 4160V power from the switchyard to the plant. This was an activity that required heightened awareness and was related to increased risk. The inspectors evaluated the following areas:

- licensed operator performance;
- the crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- the ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

Performance in these areas was compared to pre-established operator action expectations, procedural compliance, and task completion requirements. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following risk-significant systems:

- Digital Electro-hydraulic (DEH) system

The inspectors reviewed events such as where ineffective equipment maintenance had resulted in valid or invalid automatic actuations of engineered safeguards systems and independently verified the licensee's actions to address system performance or condition problems in terms of the following:

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one quarterly maintenance effectiveness samples as defined in IP 71111.12–05.

b. Findings

No findings were identified.

1R13 Maintenance Risk Assessments and Emergent Work Control (71111.13)

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the maintenance and emergent work activities affecting risk-significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- High-risk period while the PCS was in reduced inventory during 1R24;
- High-risk activity associated with the High-Pressure Turbine rotor lift;
- Emergent work to identify and troubleshoot intermittent grounds on 2400V Buses 1C, 1D, and 1E; and
- Emergent work for temporary modification and repair of P/S-0550, Master Power Supply.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each activity, the inspectors verified that risk assessments were performed as required by 10 CFR 50.65(a)(4), and were accurate and complete. When emergent work was performed, the inspectors verified that the plant risk was promptly reassessed and managed. The inspectors reviewed the scope of maintenance work, discussed the results of the assessment with the licensee's probabilistic risk analyst or shift technical advisor, and verified plant conditions were consistent with the risk assessment. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify risk analysis assumptions were valid and applicable requirements were met. Documents reviewed during this inspection are listed in the Attachment to this report.

These maintenance risk assessments and emergent work control activities constituted four samples as defined in IP 71111.13–05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- Leakage from the FLEX connection cap for the Low-Pressure Safety Injection system;
- Inability to perform the in-service valve stroke test for the Shutdown Cooling valves;
- Water and corrosion found in the Containment building floor liner leak-chase channels;
- SWS leak on Fire Protection piping in containment;
- SWS flow balancing test did not meet acceptance criteria; and
- Operability/design control of the CCW and CVCS.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee’s evaluations to determine whether the components or systems were operable. Where compensatory measures were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

This operability inspection constituted six samples as defined in IP 71111.15–05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following modification(s):

- Engineering Change 60676, Swap Qualified Incore Cables Locations with Non-Qualified Cables to Ensure 4 Qualified Circuits per Reactor Quadrant [Temporary Modification]; and
- Engineering Change 55367, Replace 1C and 1D 500MCM Feeder Cables from Startup Transformer 1-2 with 1000MCM Feeder Cables [Permanent Modification].

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system(s). The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one temporary modification sample and one permanent plant modification sample as defined in IP 71111.18–05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the following post-maintenance activities to verify that procedures and test activities were adequate to ensure system operability and functional capability:

- Completed WO package and operation post-RFO of ‘C’ PCP seal following replacement;
- Control Rod drop time testing;
- Valve stroke time testing and position verification of CV-3057, Safety Injection and Refueling Water storage tank outlet isolation valve following repairs;
- Completed WO package and satisfactory system operation of Backup Nitrogen station #5 following replacement of relief valve, RV-2279;
- Surveillance testing following 1-2 DG maintenance window;

- Completed WO package and satisfactory system operation of P-55C, 'C' Charging pump, following discharge manifold flush outlet valves replacement;
- Surveillance testing following P-54A, 'A' Containment Spray pump, maintenance window; and
- Valve stroke time testing and position verification of VOP-3007, High-Pressure Safety Injection to Reactor Coolant Loop 1A Train 1, following a condition check.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed, testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TSs, the UFSAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

This inspection constituted eight post-maintenance testing samples as defined in IP 71111.19–05.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

a. Inspection Scope

The inspectors evaluated outage activities for RFO 1R24 that began on September 16, 2015, and continued into the fourth quarter assessment period. The inspectors reviewed the Outage Risk Assessment (ORAT) and contingency plans for 1R24, prior to the shutdown, to confirm that the licensee had appropriately considered risk, industry operating experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During 1R24, the inspectors observed portions of the startup process and monitored licensee controls over the RFO activities listed below:

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the ORAT for key safety functions and compliance with the applicable TSs when taking equipment out of service;
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing;

- Installation and configuration of primary coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error;
- Controls over the status and configuration of electrical systems to ensure that TS and ORAT requirements were met, and controls over switchyard activities;
- Monitoring of decay heat removal processes, systems, and components;
- Controls to ensure that RFO work was not impacting the ability of the operators to operate the SFP cooling system;
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss;
- Controls over activities that could affect reactivity;
- Maintenance of secondary containment as required by TSs;
- Licensee fatigue management, as required by 10 CFR 26, Subpart I,
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of primary containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing; and
- Licensee identification and resolution of problems related to RFO activities.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RFO sample as defined in IP 71111.20–05.

b. Findings

No findings were identified.

1R21 Component Design Bases Inspection (71111.21)

a. Inspection Scope

In 2014, the NRC completed a Component Design Basis Inspection (CDBI) at the Palisades Nuclear Plant, documented in Inspection Report 05000255/2014008. In that inspection, the inspectors opened two separate unresolved items (URIs) related to the CCW system. The URIs are the following:

- URI 05000255/2014008-11, "Classification of CCW Piping and Components Inside the Reactor Containment Building;" and
- URI 05000255/2014008-12, "CCW System Licensing Bases."

The first URI, URI 05000255/2014008-11, is related to the ISI classification of CCW piping and components inside the reactor containment building. The inspectors performed intermittent in office inspection activities from November 2014 until October 2015. The inspection activities consisted of reviews of licensee documents, NRC requirements, NRC Safety Evaluation Reports (SERs), NRC guidance documents, and also included discussions with licensee personnel.

The Findings section below documents the conclusions reached as part of the inspectors' review. With that information, URI 05000255/2014008-11, "Classification of CCW Piping and Components inside the Reactor Containment Building," is considered closed.

The second URI, URI 05000255/2014008-12, is related to the licensing bases for the CCW system at the Palisades Nuclear Plant, and what failures the licensee is required to postulate and evaluate for the system. Although this URI is closely related to URI 05000255/2014008-11, the item is sufficiently different to warrant further inspection activities to resolve it. Therefore, URI 05000255/2014008-12, remains open.

b. Findings

(1) Failure to Identify Components Required to be Covered by the Quality Assurance Program

Introduction. The inspectors identified a finding of very-low safety significance (Green), and an associated NCV of 10 CFR, Part 50, Appendix B, Criterion II, "Quality Assurance Program," for the licensee's failure to identify all CCW structures, systems, and components (SSC), which were required to be covered by the Quality Assurance Program (i.e., be safety-related).

Description. The CCW system is discussed in Section 9.3 of the Palisades FSAR. The CCW system contains both safety-related and nonsafety-related portions. The portions of the system inside containment are nonsafety-related and are not seismically qualified. The portions of CCW inside containment were originally designed to isolate from the rest of CCW. The original FSAR, Section 9.3.1, "[Component Cooling System] Design Bases" stated:

> "The parts of the system located inside containment are isolated in the event of a [design basis accident] DBA. The remainder of the system, including pumps and heat exchangers, is located outside containment. The portion of the system located outside containment is designed to Seismic Class 1 requirements and protected from tornadoes."

The CCW system is a closed cycle system, which provides cooling water to various SSCs. Even though there are redundant pumps and heat exchangers, the system's piping is not redundant and a single pipe break or failure of the pressure boundary could result in the complete loss of CCW. One of the CCW system's safety functions is to transfer heat from the reactor and containment (post-design bases event (DBE)) to the ultimate heat sink (CCW is an intermediate system). Another safety function for the CCW system is to provide cooling to the engineered safeguards (ESF) pumps. The ESF pumps include the containment spray pumps, high-pressure safety injection pumps, and the low-pressure safety injection pumps. The CCW system also provides cooling water to the nonsafety-related charging pumps.

In addition to the system functions described above, some CCW components serve as containment isolation valves (CIVs). The original licensing bases for Palisades established the following safety-related valves as CIVs: (1) air operated valve (AOV) CV-0910 and check valve CK-CC910 on the CCW supply line to containment; and (2) AOVs CV-0911 and CV-0940 on the CCW return line from containment. Originally, the CCW portion inside containment would automatically isolate from the rest of the system during a DBE following a safety injection system (SIS) actuation. As a result, the nonsafety-related portion of CCW inside containment would isolate from the rest of the system, during a DBE, when an SIS actuation signal was present. This isolation scheme was changed in 1984 and again in 1987. The changes were documented under facility change documents FC-452-2 and FC-657, respectively. The current isolation scheme isolates the nonsafety-related CCW inside containment when a containment high pressure (CHP) signal is present, instead of an SIS signal.

In 1989 – 1990, the licensee identified single failures which could render the plant incapable of isolating the NSR portion of CCW inside containment from the safety-related portion outside of containment. As a result, the entire CCW system could be lost due to a failure of the nonsafety-related portions of CCW inside containment. The licensee was particularly concerned that a high-energy line break (HELB) inside containment could impact and breach the nonsafety-related CCW piping, resulting in the loss of CCW inventory and the loss of the entire system. This vulnerability was originally documented in Licensee Event Report (LER) 89-006, "Component Cooling Water Availability Following a HELB."

The LER addresses the licensee's concern that a single active failure of CV-0910 to close, concurrent with the failure of nonsafety-related CCW inside containment, could lead to a complete loss of the CCW system due to a loss of inventory. It should be noted that the containment integrity function would not be lost because a redundant CIV upstream of CV-0910, a check valve, could still isolate containment. In order to correct the vulnerability the licensee completed Deviation Reports D-PAL-89-061, "Post-Accident Operation of CCW System," and D-PAL-89-120, "Loss of Instrument Air – CCW System." The licensee attempted to address the vulnerabilities by demonstrating that the nonsafety-related CCW piping inside containment would not fail. They identified three potential failure mechanisms for the CCW piping inside containment: (1) missiles; (2) HELBs; and (3) earthquakes.

To address the missile failure mechanism, the licensee reviewed the Systematic Evaluation Program (SEP) Topic VI-4, "Internally Generated Missiles," which documented the NRC's evaluation of the entire CCW system (including the portion inside containment), and found it to be adequately protected against missiles. Therefore, the licensee determined that the CCW system inside containment was not vulnerable to missiles.

To address the HELB failure mechanism, the licensee prepared engineering analysis, EA-GWO 7793-01, "CCW Piping inside Containment," in 1990. The evaluation concluded that nonsafety-related portions of CCW piping inside containment were not susceptible to failure due to a HELB.

To address the seismic failure mechanism, the licensee reviewed the CCW system design. The licensee found that the CCW system piping inside containment was not seismically qualified, and therefore was susceptible to failure during an earthquake. However, since neither a Loss of Coolant Accident nor a steam line/feed line break are required to be postulated during a seismic event (because of the seismic qualifications of those systems) the licensee concluded that the plant could be brought to a safe shutdown condition without relying on the CCW system. Using the information from SEP Topic IX-3, "Station Service and Cooling Water Systems," the licensee determined: (1) reactor heat removal could be accomplished using the auxiliary feedwater system, and steam generators; and (2) primary system makeup and boration could be provided by intermittently operating the constant speed charging pumps without CCW cooling flow until CCW repairs could be made. It is important to note, that at the time of this evaluation (1989-1990), the charging pumps were categorized as safety-related.

Based on the above, the licensee decided to credit the nonsafety-related CCW system inside containment to remain intact during a HELB (not vulnerable to HELB damage), and documented this new licensing basis in their FSAR. This change was processed on August 21, 1990, as FSAR change request 5-39-R11-391. In effect, the licensee added a safety function to the CCW piping inside containment by now relying on it to shut down the reactor and maintain it in a safe shutdown condition during and following a DBE, which causes a HELB inside containment, given the single active failure of CIV CV-0910. This change was made incorrectly because the licensee failed to either reclassify the CCW piping inside containment as safety-related when it was credited to perform a safety function, or obtain prior NRC approval to credit the nonsafety-related piping to perform a safety function.

In 1999, the licensee made changes to containment penetrations MZ-14 (CCW supply) and MZ-15 (CCW return). The changes were made using the conclusion that CCW piping inside containment was not considered susceptible to failure (as discussed above). These penetrations are part of the containment isolation system, which is discussed in Section 6.7 of the FSAR. Originally, each containment penetration credited two in-series CIVs and the CCW system piping inside containment was not credited. The containment penetrations were then changed to credit one CIV per penetration (check valve CK-CC910 and AOV CV-0911), and the CCW system piping inside containment because it was no longer believed to be susceptible to failure. In effect, the licensee added an additional safety function to the CCW piping inside containment by now relying on it to form a part of the containment boundary. This change was evaluated under 50.59 Unreviewed Safety Question Evaluation SDR-99-0884. The change was made incorrectly because the licensee failed to either reclassify the CCW piping inside containment as safety-related when it was credited it to perform an additional safety function, or obtain prior NRC approval to credit the nonsafety-related piping to perform a safety function. It is important to note that: (1) the instrument air supplied to the CIV AOV CV-0911 is nonsafety-related; (2) CV-0911 is a fail open valve; (3) CV-0911 has an air accumulator rated for four hours; but it is also nonsafety-related; (4) the CCW piping inside containment credited to form a part of the containment barrier is not seismically qualified; and (5) once the containment penetration changes were made the local leak rate testing requirement of the remaining credited CCW CIVs was removed.

Title 10 CFR 50.2, in part, defines safety-related SSCs as those SSCs that are relied upon to remain functional during and following design basis events to assure: (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents, which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 100.11. This definition of safety-related is also included in the licensee's FSAR under Section 5.2.2.8.1. In addition, 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program," requires in part, that licensees identify the SSCs to be covered by the Quality Assurance Program (i.e., safety-related components).

Based on the above, the inspectors determined the licensee has incorrectly made changes to the plant to credit nonsafety-related SSCs with a safety-related function without reclassifying them as safety-related. The nonsafety-related SSCs and their safety-related functions are the following:

- Given a postulated single active failure of CIV CV-0910, the nonsafety-related CCW piping inside containment is credited with a safety-related function to maintain its pressure boundary (remain intact) following a DBE (excluding a seismic event). This function is required in order for the CCW system to perform its safety-related reactor and containment heat removal functions needed to: (1) shut down the reactor and maintain it in a safe shutdown condition; and (2) prevent or mitigate the consequences of accidents.

- The nonsafety-related CCW piping inside containment is credited to maintain its pressure boundary (remain intact) following a DBE to provide a safety-related containment integrity barrier (the first of two barriers for containment penetrations MZ-14 and MZ-15) to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Analysis. The inspectors determined the licensee failed to identify all CCW SSCs, which were required to be covered by the Quality Assurance Program, i.e., be safety-related SSCs in accordance with 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program." As a result, the licensee incorrectly credited nonsafety-related CCW components to remain functional during and following a DBE. This was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, since nonsafety-related components are not maintained to the same design, fabrication, construction, and testing quality standards as safety-related SSCs, these nonsafety-related components could not be relied upon to perform their functions during a DBE.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "SDP," 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," issued June 19, 2012, and answered "yes" to Question A.1. The finding screened as having very low safety significance (Green) because, although it was a deficiency affecting the design or qualification of a mitigating SSC, the SSC maintained its operability. Specifically the licensee evaluated the CCW system's operability as part of corrective actions CR-PLP-2015-01872, and CR-PLP-2015-05468, and concluded the system would still be capable of performing its function.

The inspectors did not identify a cross-cutting aspect associated with this finding. Based on the dates (1989 and 1999) when the modifications were implemented, the inspectors determined the finding was not representative of current performance.

Enforcement. Title 10 CFR, Part 50, Appendix B, Criterion II, "Quality Assurance Program," requires, in part, that the licensee shall identify the SSCs to be covered by the Quality Assurance Program.

Title 10 CFR, Part 50, Appendix B, “Introduction,” states, in part, that nuclear power plants and fuel reprocessing plants include SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. This appendix establishes quality assurance requirements for the design, manufacture, construction, and operation of those SSCs. The pertinent requirements of this appendix apply to all activities affecting the safety-related functions of those SSCs, these activities include designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, and modifying. As used in this appendix, “quality assurance” comprises all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. Quality assurance includes quality control, which comprises those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements.

Title 10 CFR Part 50.2, “Definitions,” states, in part, that safety-related SSCs means those SSCs that are relied upon to remain functional during and following DBEs to assure: (1) the integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in § 50.34(a)(1) or § 100.11 of this chapter, as applicable.

Contrary to the above, since August 21, 1990, the licensee failed to identify all the CCW SSCs which were required to be covered by the Quality Assurance Program. Quality assurance requirements apply to all activities affecting the safety-related functions of SSCs and comprise all those planned and systematic actions necessary to provide adequate confidence that a SSC will perform satisfactorily in service.

Specifically, the portions of CCW inside containment meet the 10 CFR 50.2 definition of “Safety-Related SSCs,” and are required to be covered by the Quality Assurance Program because they are credited to remain functional during and following design basis events to ensure: (1) the capability to shut down the reactor and maintain it in a safe shutdown condition; and (2) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guidelines. In particular, the nonsafety-related portions of CCW inside containment: (1) are part of the pressure boundary which ensure CCW inventory is maintained so the safety-related portions of CCW can perform their function; and (2) form part of the containment isolation system as one of the two containment barriers, which ensure containment can perform its safety-related function.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy. The violation was entered into the licensee’s CAP as CR-PLP-2015-01872 and CR-PLP-2015-05468. As part of their immediate corrective actions, the licensee performed operability determinations which concluded the system would still be capable of performing its function. **(NCV 05000255/2015004-02; Failure to Identify Components Required to be Covered by the Quality Assurance Program)**

(2) Failure to Provide Bases to Determine Changes Did Not Involve Unreviewed Safety Questions

Introduction. The inspectors identified a Severity Level (SL) IV, NCV of 10 CFR 50.59, "Changes, Tests, and Experiments," for the licensee's failure to maintain records of written safety evaluations which provide the bases for concluding the nonsafety-related portions of the CCW system inside containment could be credited to perform their function during and following a DBE, and that the change would not result in an unreviewed safety question.[1]

Description. The CCW system is discussed in Section 9.3 of the Palisades FSAR. The CCW system contains both safety-related and nonsafety-related portions. The portions of the system inside containment are classified as nonsafety-related, and are not seismically qualified. The portions of CCW inside containment were originally designed to automatically isolate from the rest of CCW. The original FSAR, Section 9.3.1, "[Component Cooling System] Design Bases" stated:

> "The parts of the system located inside containment are isolated in the event of a DBA. The remainder of the system, including pumps and heat exchangers, is located outside containment. The portion of the system located outside containment is designed to Seismic Class 1 requirements and protected from tornadoes."

The CCW system is a closed cycle system which provides cooling water to various SSCs. Although CCW has redundant pumps and heat exchangers, the system's piping is not redundant and a single pipe break or failure of the pressure boundary could result in the complete loss of CCW. Safety functions for the CCW system include: (1) transferring heat from the reactor and containment to the ultimate heat sink (serving as an intermediate loop); (2) providing cooling to the ESF pumps; and (3) serving as part of the containment isolation scheme.

The original licensing bases established a number of CCW SR valves as CIVs. For the supply line to containment the valves are: AOV CV-0910 and check valve CK-CC910. For the return line the valves are: AOVs CV-0911 and CV-0940. These CIVs would automatically close following a SIS actuation. As a result, the nonsafety-related portions of CCW inside containment would automatically isolate from the rest of the system, during a DBE, when an SIS actuation signal was present. This isolation scheme was changed in 1984, and again in 1987. The changes were documented under facility change documents FC-452-2 and FC-657, respectively. The current isolation scheme no longer isolates the nonsafety-related CCW inside containment on a SIS, but instead isolates it on a CHP signal.

In 1989-1990, the licensee identified single failures which could render the plant incapable of isolating the nonsafety-related portions of CCW inside containment from the safety-related portions outside of containment. As a result, the entire CCW system could be lost if any of the nonsafety-related portions of CCW inside containment failed. This vulnerability was originally documented in LER 89-006, "CCW Availability Following a HELB." To resolve the concern, the licensee performed a series of evaluations and

[1] It is important to note that until 1999 the 10 CFR Part 50 Section 59 rule was different than the current version. As a result; the inspectors evaluated the licensee's activities against the version in effect at the time the changes in question were performed.

concluded the nonsafety-related portions of CCW inside containment were not vulnerable to failure (except following a seismic event). As a result, the licensee decided to credit the nonsafety-related portions of CCW inside containment as not being vulnerable to failure during a DBE (except during a seismic event). This was a new licensing bases for Palisades, and an FSAR update was processed under FSAR Change No. 5-39-RII-391 on August 21, 1990. The current version of FSAR Section 9.3 (Revision 29) discusses the HELB concern regarding CCW inside containment in Section 9.3.2.3.

In July 27, 1990, the FSAR change was evaluated for 50.59 implications under Unreviewed Safety Question Evaluation 90-1063. The evaluation concluded no NRC-approval was required prior to implementing the change. The evaluation, however, failed to recognize the change could increase the probability of malfunction of equipment important to safety. Specifically, the licensee answered “No” to Question 3 of Evaluation 90-1063, Section I, which asked, “Will the probability of malfunctions of equipment important to safety be increased?” The inspectors determined that, as a result of the change, in order for the safety-related portions of CCW to perform their design bases function, the pressure boundary of the nonsafety-related/non-seismic CCW portions inside containment (i.e., piping, relief valve, heat exchangers, etc.) must remain intact during and following a DBE. In other words, the change made the probability of malfunction of the safety-related portions of CCW dependent on the proper functioning of the nonsafety-related portions of CCW inside containment. The nonsafety-related portions are not maintained, evaluated or tested to the same quality assurance standards as the safety-related portions. The inspectors concluded the licensee failed to provide the bases for determining the change would not result in an unreviewed safety question as defined in 10 CFR 50.59(a)(2). The 50.59 rule required licensees to submit a license amendment request for changes deemed to be unreviewed safety questions.

In addition to the changes discussed above, in 1999, the licensee re-classified the containment penetrations associated with CCW. The penetrations were MZ-14 and MZ-15, CCW supply and return from containment, respectively. These penetrations are part of the containment isolation system, which is discussed in Section 6.7 of the FSAR. The new containment isolation scheme relies on the nonsafety-related pressure boundary of CCW inside containment, and one CIV per penetration (CK-CC910 and AOV CV-0911). In July 22, 1999, this change was evaluated under 10 CFR 50.59, Unreviewed Safety Question Evaluation SDR-99-0884. The evaluation also concluded no NRC-approval was required prior to implementing the change. However, similar to the discussion above, the licensee failed to recognize the increased probability of malfunction of the safety-related containment isolation system in Section I.3 of the evaluation. Specifically, the change made the containment isolation design bases safety function dependent on the nonsafety-related/non-seismic CCW SSCs' pressure boundary inside containment, thereby requiring these nonsafety-related components to remain functional during and following a DBE. Each penetration went from having two safety-related components as the containment isolation barrier to one safety-related component and one nonsafety-related barrier. As described previously, nonsafety-related components are not maintained, evaluated or tested to the same quality assurance standards as the safety-related portions. These quality standards ensure safety-related SSCs can be relied upon to perform their function during and following a DBE. The inspectors concluded the licensee failed to provide the bases for determining the change would not result in an unreviewed safety question as defined in 10 CFR 50.59(a)(2). The 50.59 rule required licensees to submit a license amendment request for changes deemed to be unreviewed safety questions.

The inspectors concluded the changes made in 1990 and 1999, violated the requirements of 10 CFR 50.59. The underlying technical concern associated with this violation resulted in a Green finding and an additional NCV of 10 CFR Part 50, Appendix B, Criterion II, "Quality Assurance Program." The finding was documented in this report under 05000255/2015004-02, "Failure to Identify Components Required to be Covered by the Quality Assurance Program." The licensee documented the inspectors' concerns in the CAP as CR-PLP-2015-01872 and CR-PLP-2015-05468.

Analysis. The inspectors determined the licensee failed to maintain records of written safety evaluations which provided the bases for the conclusion that nonsafety-related portions of CCW could be credited to perform their function during and following a DBE, and that this change would not result in an unreviewed safety question. This was contrary to 10 CFR 50.59(b)(1), and was a violation of regulatory requirements. Specifically, the licensee did not provide the bases to explain why crediting the nonsafety-related portions of CCW inside containment to perform their function would not result in an increase in the probability of malfunction of the safety-related CCW or safety-related containment isolation systems. A bases is essential for crediting the nonsafety-related SSCs because they are not maintained to the same design, fabrication, construction, and testing quality standards as safety-related SSCs.

The violation of 10 CFR 50.59 was determined to be more than minor because the inspectors could not reasonably determine that the changes would not have ultimately required NRC prior approval. Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the SDP because they are considered to be violations that potentially impede or impact the regulatory process. This violation is associated with a finding that has been evaluated by the SDP and communicated with an SDP color reflective of the safety impact of the deficient licensee performance. The SDP, however, does not specifically consider the regulatory process impact. Thus, although related to a common regulatory concern, it is necessary to address the violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

In this case, the inspectors determined the underlying technical concern resulted in a Green finding. The details of the finding and its significance determination were documented in this report as 05000255/2015004-02, "Failure to Identify Components Required to be Covered by the Quality Assurance Program."

In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation was categorized as a SL IV because the resulting changes were evaluated by the SDP as having very low safety significance (i.e., Green finding).

Cross-cutting aspects are not assigned to traditional enforcement violations.

Enforcement. The version of the 10 CFR 50.59 rule, in effect, in 1990 and 1999, was different than the current version. At the time, 10 CFR 50.59(b)(1) required, in part, that the licensee shall maintain records of tests and experiments carried out pursuant to paragraph (a) of the section. These records must include a written safety evaluation which provides the bases for the determination that the change, test, or experiment does not involve an unreviewed safety question. In addition, 10 CFR 50.59(c)(2) stated, in part, that a licensee desiring to make a change in the facility or the procedures described in the safety analysis report or to conduct tests or experiments not described in the safety analysis report, which involve an unreviewed safety question or a change in technical specifications, shall submit an application for amendment of his license pursuant to 10 CFR 50.90. In accordance with 10 CFR 50.59(a)(2), a change was deemed to involve an unreviewed safety question if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased.

Contrary to the above, on July 27, 1990, and July 22, 1999, the licensee failed to maintain records of written safety evaluations which provided the basis for determining the changes did not involve an unreviewed safety question, as defined in 10 CFR 50.59(a)(2), and would not require a license amendment pursuant to 10 CFR 50.90. Specifically:

- Palisades' Unreviewed Safety Question Evaluation 90-1063, Section I.3, failed to provide the basis for determining there was no increased probability of malfunction of the safety-related portions of CCW, which are equipment important to safety. Particularly, the change made the ability of the safety-related portions of CCW to perform their function dependent on the ability of the nonsafety-related portions of CCW inside containment to maintain their pressure boundary; and

- Palisades' Unreviewed Safety Question Evaluation 99-0884, Section I.3, failed to provide the basis for determining there was no increased probability of malfunction of the safety-related containment isolation system, which is equipment important to safety. Particularly, the change made the ability of the safety-related containment isolation system to perform its function dependent on the ability of nonsafety-related portions of CCW inside containment to maintain their pressure boundary.

The nonsafety-related CCW components inside containment are not maintained to the same design, fabrication, construction, and testing quality standards as safety-related components. Quality assurance standards ensure safety-related SSCs can be relied upon to perform their function during and following a DBE. In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation of 10 CFR 50.59 is classified as a SL IV Violation. The Enforcement Policy classifies violations of 10 CFR 50.59 as SL IV if the resulting conditions are evaluated by the SDP as having a very low safety significance (i.e., Green).

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was a SL IV violation and was entered into the licensee's CAP as CR-PLP-2015-01872 and CR-PLP-2015-05468. As part of their immediate corrective actions, the licensee performed operability determinations which concluded the system would still be capable of performing its function.
(NCV 05000255/2015004-03; Failure to Provide Bases to Determine Changes Did Not Involve Unreviewed Safety Questions)

(3) Failure to Perform a Required 50.59 Evaluation for Declassification of the Chemical and Volume Control System

Introduction. The inspectors identified a SL IV, NCV of 10 CFR 50.59, "Changes, Tests, and Experiments," and an associated finding of very-low safety significance (Green) for the licensee's failure to maintain a record of a change in the facility which includes a written evaluation that provides the bases for the determination that the change did not require a license amendment. Specifically, the licensee failed to maintain a record of the declassification of the Chemical and Volume Control System (CVCS) from safety-related to nonsafety-related, which includes a written evaluation that provides the bases for the determination that the change did not require a license amendment pursuant to paragraph (c)(2)(ii) of 10 CFR 50.59, because it did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of the system as previously evaluated in Sections 5.1, 5.2, and 1.8.5 of the FSAR.

Description. FSAR Section 5.1 discusses Palisade's compliance with the NRC General Design Criteria (GDC) that were in place at the time the plant was licensed. FSAR Section 5.1.2.2, "Criterion 2 – Design Bases for Protection Against Natural Phenomena," describes Palisades' compliance with GDC 2. It states that, "this criterion has been met by designing, fabricating and erecting those SSCs important to safety to withstand the effects of extraordinary natural phenomena." A seismic event is one of the natural phenomena events discussed in GDC 2, which the licensee is required cope with. At Palisades, the CVCS is one of the systems credited to achieve and maintain a safe shutdown condition following a seismic event, and hence it is required to meet GDC 2 for Palisades.

Originally, the CVCS was considered a Consumers Design Class 1 system. As discussed in FSAR Section 5.2.1.2, "Original Palisades Design Review," Consumers Design Class is a combination of Safety Class and Seismic Class. In FSAR Section 5.2.2.1, "Design – Class 1," describes Consumers Design Class 1 by stating that, "Class 1 systems and components were designed for functional dependability following an earthquake by using the load combinations in Section 5.10.1. Class 1 systems and components are always Seismic Category I equivalents in current design practice, however, they may be equivalent to ASME Boiler and Pressure Vessel (B&PV) Class 1, 2 or 3. Class 1 systems could also be Safety Class 1, 2 or 3 per American National Standards Institute (ANSI) N18.2-1973. Table 5.2-3 identifies systems' classification and industrial design codes utilized." Therefore, the seismic classification of the CVCS was originally considered equivalent to Seismic Category I.

In December of 1980, the NRC initiated Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Nuclear Plants," because equipment in nuclear plants for which construction permit applications had been docketed before about 1972 had not been reviewed according to the then-current (1980-1981), licensing criteria for seismic qualification of equipment. Therefore, the seismic adequacy of the equipment in these older plants and the equipment's ability to survive and function in the event of a safe shutdown earthquake (SSE) was in question. Palisades was one of the plants for which the concern existed.

On May 19, 1995, Palisades submitted a report, "Report of Seismic Qualification Utility Group (SQUG) Assessment at Palisades Nuclear Plant for the Resolution of USI A-46," to resolve NRC USI A-46 for the seismic qualification of equipment at the plant. The report used the SQUG Generic Implementation Procedure (GIP), endorsed by NRC Generic Letter 87-02, Supplement 1, to identify the preferred paths to be used in accomplishing safe shutdown functions following a SSE. Once the preferred paths were identified, specific equipment in the safe shutdown paths was evaluated using the SQUG GIP. The CVCS, also known as the charging system, was credited as part of the preferred paths and the seismic qualification of CVCS equipment was verified in the process. Subsequently, the NRC issued a SER on September 25, 1998, documenting the staff's conclusions on their review of the Palisades' SQUG Report and closure of the NRC USI A-46 review for Palisades.

The information on resolution of USI A-46 for Palisades was approved for incorporation into the FSAR in September of 2003, and is discussed in Section 1.8.5 of the FSAR. That section of the FSAR describes the credit taken by the licensee, since at least 1995, for the CVCS as one of the systems used to attain the safe shutdown condition following a seismic event. Specifically, it describes the design function of the charging system (CVCS) to provide a flow path to, "ensure the maintenance of inventory and boron concentration in the primary coolant system while the plant is shutdown following a seismic event." This CVCS design function is further described in the Palisades SQUG Report, which was approved by the NRC SER, and is a part of the current licensing basis for Palisades. The SQUG Report describes the CVCS as part of the safe shutdown paths to accomplish the reactor reactivity control and reactor inventory control safe shutdown functions following a SSE. Therefore, the CVCS, including the charging pumps, has a design function to mitigate the consequences of a seismic event.

Since at least 1995, when the CVCS was credited to mitigate the consequences of a seismic event for resolution of USI A-46, it has been required to meet GDC 2 for Palisades. At that time, the system was classified as a Consumers Design Class 1 system and was considered equivalent to a Seismic Category I system. In addition, the seismic qualification of equipment which formed part of the system had also been verified through the SQUG methodology. The system was also considered safety-related and subject to the QA requirements of 10 CFR 50, Appendix B. Therefore, at that time, the CVCS met the requirements of GDC 2 for Palisades by having the qualifications described above.

In 2003, the licensee decided to declassify most portions of the CVCS, including the charging pumps, from a safety-related classification to a nonsafety-related classification. This was done, in part, to resolve an operable but degraded condition of some of the piping in the system. In September 22, 2003, the licensee completed a 10 CFR 50.59 screening, SDR-03-1073, "CVCS Declassification," and concluded that the change did not involve adverse effects and therefore, did not require a 10 CFR 50.59 evaluation. The licensee then proceeded with the change and declassified most portions of the system. The declassification removed all 10 CFR Part 50, Appendix B, QA requirements and ASME Code requirements for most portions of the system. It also removed the seismic qualification requirements for most portions of the system, including the charging pumps. Therefore, the CVCS qualification requirements for meeting GDC 2, as described above, were either entirely removed or significantly reduced.

Through a review of the 10 CFR 50.59 screening, the inspectors noted that the CVCS function for mitigating the consequences of a seismic event was not identified as a design function affected by the change. Therefore, the licensee did not review all CVCS design functions when determining the change did not involve adverse effects.

Per Section 4.2.1 of Nuclear Energy Institute (NEI) 96-07, Revision 1, "Guidelines for 10 CFR 50.59 Implementation," endorsed by NRC Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," a change that decreases the reliability of a function whose failure could initiate an accident would be considered to adversely affect a design function and would screen in and require a 10 CFR 50.59 evaluation. It also states that, "changes that would relax the manner in which Code requirements are met for certain SSCs should be screened for adverse effects on design function."

Since the declassification of most portions of the CVCS was a change that, among other things, removed QA and Code requirements, the change was a relaxation of Code requirements that could be reasonably viewed as an adverse change that could decrease the reliability of the system. In addition, since the seismic qualification requirements for the system were also removed, the change could be reasonably viewed as an adverse change because the system was no longer required to be qualified for the external event (seismic event) it is credited for to get the plant to a safe shutdown condition. Had the licensee identified the design function of the CVCS to mitigate the consequences of a seismic event, they would have had to evaluate whether the changes made to the system adversely affected the design function. As described above, the inspectors determined the changes made to the system could be reasonably viewed as adverse changes, and therefore the licensee was required to perform a 10 CFR 50.59 evaluation. It is worth noting that although the plant has other means of getting the plant to a safe shutdown condition following a SSE, the equipment in those paths was not seismically verified as part of the resolution for USI A-46, and cannot be credited following a SSE.

The inspectors reviewed the requirement in 10 CFR 50.59(c)(2)(ii) for determining whether a change results in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR (as updated). The inspectors also reviewed the guidance in NEI 96-07 for making that determination. Section 4.3.2 of NEI 96-07 states that, "departures from the design, fabrication, construction, testing and performance standards as outlined in the GDC (Appendix A to Part 50) are not compatible with a "no more than minimal increase" standard." Since the declassification of the system resulted in a departure from the licensee's original compliance with GDC 2, the change could be reasonably viewed as resulting in a more than a minimal increase in the likelihood of occurrence of a malfunction of the important to safety CVCS.

In addition, Section 4.3.2 of NEI 96-07 states that, "changes in design requirements for earthquakes, tornadoes, and other natural phenomena should be treated as potentially affecting the likelihood of malfunction." Since the licensee changed the design requirements of the CVCS for a seismic event, the change could again be reasonably viewed as a change resulting in a more than a minimal increase in the likelihood of the system's malfunction.

Based on the information described above, the inspectors determined that the licensee failed to provide a written evaluation which provided the bases for the determination that the change, reclassification of the CVCS, did not require a license amendment pursuant to 10 CFR 50.59(c)(2). The inspectors could not reasonably determine that the change made would not have ultimately required NRC prior approval, because it could be reasonably viewed as a change which caused a more than a minimal increase in the likelihood of occurrence of a malfunction of the CVCS.

The licensee captured the inspectors' concern in the CAP as condition report CR-PLP-2015-01873. The licensee's immediate corrective actions included a review of the system and determination that there was reasonable assurance that it could perform its function because its configuration had not been altered to remove features which would allow its operation following a seismic event.

Analysis. Violations of 10 CFR 50.59 can be dispositioned using both the traditional enforcement process and the SDP because in addition to being violations that potentially impede or impact the NRC's ability to perform its regulatory oversight function, they may be associated with underlying technical issues that have potential safety significance. First, the underlying technical issue is evaluated to determine whether it's a finding, and if so, an SDP color reflective of its safety significance is determined. Then, because the SDP does not specifically consider the regulatory process impact, the finding's safety significance is used to assign a traditional enforcement SL using Section 6.1, "Reactor Operations," of the NRC Enforcement Policy. Thus, although related to a common regulatory concern, it is necessary to address the traditional enforcement violation and finding using different processes to correctly reflect both the regulatory importance of the violation and the safety significance of the associated finding.

To evaluate the issue of concern using the SDP process, the inspectors first determined that the change to the CVCS safety classification was contrary to 10 CFR 50.59(d)(1) and was a performance deficiency. Specifically, the licensee failed to provide the bases for the determination that the declassification of the CVCS, the change in its seismic design requirements, and the removal of quality standards for design, fabrication, and testing of the system did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of the system and did not require a license amendment. The inspectors determined that the performance deficiency was more than minor because, if left uncorrected, it would become a more significant safety concern. Specifically, the failure to maintain seismic qualification requirements and other quality requirements which control the design, fabrication, and testing quality standards for the CVCS could result in changes being implemented which affect the ability of the charging system to successfully respond to and mitigate the effects of a seismic event. The inspectors concluded this finding was associated with the Mitigating Systems cornerstone.

The inspectors also determined that the issue of concern described above was a traditional enforcement violation because it was a violation of 10 CFR 50.59 and was considered to potentially impede or impact the NRC's ability to perform its regulatory oversight function. The traditional enforcement violation was determined to be more than minor because the inspectors could not reasonably determine that the change made, reclassification of the CVCS, would not have ultimately required NRC prior approval.

The inspectors determined the finding described above could be evaluated using the SDP in accordance with IMC 0609, "SDP," issue date April 29, 2015, Attachment 0609.04, "Initial Characterization of Findings," issue date June 19, 2012. Because the finding impacted the Mitigating Systems cornerstone, the inspectors screened the finding through IMC 0609 Appendix A, "The SDP for Findings At-Power," issue date June 19, 2012, using Exhibit 2, "Mitigating Systems Screening Questions," and answered "yes" to Question A.1. The finding screened as having very low safety significance (Green) because although it affected the design or qualification of the CVCS, it did not result in the loss of functionality of the CVCS. Specifically, the licensee determined that although the CVCS components were declassified, no physical changes or modifications had been implemented, which affected the SSCs' ability to perform its mitigating function following a seismic event.

In accordance with Section 6.1.d.2, "Reactor Operations," of the NRC Enforcement Policy this violation is categorized as SL IV because the resulting changes were evaluated by the SDP as having very low safety significance (i.e., green finding).

The inspectors did not identify a cross-cutting aspect associated with the finding because the finding was not representative of current performance. Specifically, the declassification of the CVCS was completed in 2003.

Enforcement. Title 10 CFR Part 50.59, "Changes, Tests, and Experiments," Section (d)(1) requires the licensee to maintain records of changes in the facility, of changes in procedures, and of tests and experiments made pursuant 10 CFR 50.59(c). These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment pursuant to paragraph (c)(2) of the section.

Title 10 CFR Part 50.59 (c)(2)(ii) states, in part, that a licensee shall obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing a proposed change, test, or experiment if the change, test, or experiment would result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR (as updated).

FSAR Section 5.1.2.2, "Criterion 2 – Design Bases for Protection Against Natural Phenomena," describes Palisades' compliance with draft GDC 2. It states that, "this criterion has been met by designing, fabricating and erecting those SSCs important to safety to withstand the effects of extraordinary natural phenomena."

FSAR Section 5.2.2.1, "Design – Class 1" states that, "Class 1 systems and components were designed for functional dependability following an earthquake by using the load combinations in Section 5.10.1. Class 1 systems and components are always Seismic Category I equivalents in current design practice, however, they may be equivalent to ASME B&PV Class 1, 2 or 3. Class 1 systems could also be Safety Class 1, 2 or 3 per ANSI N18.2-1973. Table 5.2-3 identifies systems' classification and industrial design codes utilized. "FSAR Section 1.8.5, "USI (NUREG-0410)," describes a design function of the charging system [CVCS] to provide a flow path, "to ensure the maintenance of inventory and boron concentration in the primary coolant system while the plant is shutdown following a seismic event.

Contrary to the above, since September 22, 2003, the licensee failed to maintain a record of a change in the facility which includes a written evaluation that provides the bases for the determination that the change did not require a license amendment pursuant to paragraph (c)(2) of 10 CFR 50.59. Specifically, the licensee failed to maintain a record of the declassification of the CVCS from safety-related to nonsafety-related, which included a written evaluation that provided the bases for the determination that the change did not require a license amendment pursuant to paragraph (c)(2)(ii) of 10 CFR 50.59 because it did not result in more than a minimal increase in the likelihood of occurrence of a malfunction of the system as previously evaluated in Sections 5.1, 5.2 and 1.8.5 of the FSAR. In particular, by declassifying the CVCS and removing seismic qualification and quality requirements, the system was no longer subject to the original and more stringent requirements that ensured any maintenance and/or changes were performed in a quality manner so as to maintain the CVCS design function for mitigating a seismic event.

This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was a SL IV violation and was entered into the licensee's CAP as CR-PLP-2015-01873. As part of the immediate corrective actions, the licensee performed a review of the system, and determined there was reasonable assurance the CVCS could perform its function because its configuration had not been altered to remove features which would allow its operation following a seismic event. **(NCV 05000255/2015004-04; Failure to Perform a Required 50.59 Evaluation for Declassification of the CVCS).**

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- RT-8C, left train Engineered Safeguards system integrated testing (routine);
- RT-36, Containment leak test (routine);
- RO-12, Containment High Pressure and Spray test (routine);
- RT-191, Low Power Physics testing (routine);
- QO-21A, 'A' Auxiliary Feedwater pump in-service testing (IST); and
- RO-19, Control Rod Drive position verification test (routine).

The inspectors observed in-plant activities and reviewed procedures and associated records to determine the following:

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, sufficient to demonstrate operational readiness, and consistent with the system design basis;
- was plant equipment calibration correct, accurate, and properly documented;
- were as-left setpoints within required ranges, and was the calibration frequency in accordance with TSs, the UFSAR, plant procedures, and applicable commitments;
- was measuring and test equipment calibration current;
- was the test equipment used within the required range and accuracy and were applicable prerequisites described in the test procedures satisfied;
- did test frequencies meet TS requirements to demonstrate operability and reliability;

- were tests performed in accordance with the test procedures and other applicable procedures;
- were jumpers and lifted leads controlled and restored where used;
- were test data and results accurate, complete, within limits, and valid;
- was test equipment removed following testing;
- where applicable for inservice testing activities, was testing performed in accordance with the applicable version of Section XI of the ASME Code, and were reference values consistent with the system design basis;
- was the unavailability of the tested equipment appropriately considered in the performance indicator (PI) data;
- where applicable, were test results not meeting acceptance criteria addressed with an adequate operability evaluation, or was the system or component declared inoperable;
- where applicable for safety-related instrument control surveillance tests, was the reference setting data accurately incorporated into the test procedure;
- was equipment returned to a position or status required to support the performance of its safety function following testing;
- were all problems identified during the testing appropriately documented and dispositioned in the licensee's CAP;
- where applicable, were annunciators and other alarms demonstrated to be functional and were annunciator and alarm setpoints consistent with design documents; and
- where applicable, were alarm response procedure entry points and actions consistent with the plant design and licensing documents.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted five routine surveillance testing samples and one in-service test sample as defined in IP 71111.22, Sections-02 and-05.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (IP 71114.04)

a. Inspection Scope

The regional inspectors performed an in-office review of the latest revisions to the Emergency Plan, Emergency Action Levels (EALs).

The licensee transmitted the Emergency Plan and EAL revisions to the NRC pursuant to the requirements of 10 CFR, Part 50, Appendix E, Section V, "Implementing Procedures." The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes, therefore, this revision is subject to future inspection.

This EAL and Emergency Plan Changes inspection constituted one sample as defined in Inspection Procedure 71114.04.

b. Findings

No findings were identified.

2. RADIATION SAFETY

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted one complete sample as defined in IP 71124.01-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of radiation protection program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

The inspectors determined if there have been changes to plant operations since the last inspection that may result in a significant new radiological hazard for onsite workers or members of the public. The inspectors evaluated whether the licensee assessed the potential impact of these changes and has implemented periodic monitoring, as appropriate, to detect and quantify the radiological hazard.

The inspectors reviewed the last two radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys were appropriate for the given radiological hazard.

The inspectors conducted walkdowns of the facility, including radioactive waste processing, storage, and handling areas to evaluate material conditions and performed independent radiation measurements to verify conditions.

The inspectors selected the following radiologically risk-significant work activities that involved exposure to radiation:

- 1R24 Radiography activities;
- Refuel Project: Incore Instrumentation (ICI) installation/removal;
- Pressurizer Spray Control Valves, CV-1057 & CV-1059;
- Refuel Project: Replace proximity switches on Reactor side Tilt Machine and remove debris for Reactor Tilt Pit; and
- Steam Generator inspections and repairs (Primary side).

For these work activities, the inspectors assessed whether the pre-work surveys performed were appropriate to identify and quantify the radiological hazard and to establish adequate protective measures. The inspectors evaluated the radiological survey program to determine if hazards were properly identified, including the following:

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials (This evaluation may include licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel.);
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors evaluated the licensee's program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected various containers holding non-exempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR, Part 20.1904, "Labeling Containers," or met the requirements of 10 CFR 20.1905(g), "Exemptions To Labeling Requirements".

The inspectors reviewed the following radiation work permits (RWPs) used to access high radiation areas and evaluated the specified work control instructions or control barriers:

- RWP 20150487: 1R24 Radiography Activities;
- RWP 20150429: Refuel Project: ICI Installation/Removal;
- RWP 20150468: Pressurizer Spray Control Valves CV-1057 & CV-1059;
- RWP 20150431: Refuel Project: Replace Proximity Switches on Reactor Side Tilt Machine and Remove Debris for Reactor Tilt Pit; and
- RWP 20150454: Steam Generator Inspections and Repairs (Primary Side).

For these RWPs, the inspectors assessed whether allowable stay times or permissible dose (including from the intake of radioactive material) for radiologically significant work under each RWP were clearly identified. The inspectors evaluated whether electronic personal dosimeter alarm set-points were in conformance with survey indications and plant policy.

The inspectors reviewed selected occurrences where a worker's electronic personal dosimeter noticeably malfunctioned or alarmed. The inspectors evaluated whether workers responded appropriately to the off-normal condition. The inspectors assessed whether the issue was included in the CAP, and dose evaluations were conducted as appropriate.

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee's means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

No findings were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiological control area and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures and whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the type(s) of radiation present.

The inspectors reviewed the licensee's criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee's procedures and records to verify that the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a de facto "release limit" by altering the instrument's typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

The inspectors evaluated whether any transactions, since the last inspection, involving nationally tracked sources were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors evaluated ambient radiological conditions (e.g., radiation levels or potential radiation levels) during tours of the facility. The inspectors assessed whether the conditions were consistent with applicable posted surveys, RWPs, and worker briefings.

The inspectors evaluated the adequacy of radiological controls, such as required surveys, radiation protection job coverage (including audio and visual surveillance for remote job coverage), and contamination controls. The inspectors evaluated the licensee's use of electronic personal dosimeters in high-noise areas as high-radiation area monitoring devices.

The inspectors assessed whether radiation monitoring devices were placed on the individual's body consistent with licensee procedures. The inspectors assessed whether the dosimeter was placed in the location of highest expected dose or that the licensee properly employed an NRC-approved method of determining effective dose equivalent.

The inspectors reviewed the application of dosimetry to effectively monitor exposure to personnel in high-radiation work areas with significant dose rate gradients.

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures:

- RWP 20150429; Refuel Project: ICI Installation/ Removal;
- RWP 20150468; Pressurizer Spray Control Valves CV-1057 & CV-1059;
- RWP 20150431; Refuel Project: Replace Proximity Switches on Reactor Side Tilt Machine and Remove Debris for Reactor Tilt Pit; and
- RWP 20150454; Steam Generator Inspections and Repairs (Primary Side).

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potential for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, and reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary high-efficiency particulate air ventilation system operation.

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (i.e., nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors examined the posting and physical controls for selected high-radiation areas and very-high radiation areas to verify conformance with the occupational performance indicator.

b. Findings

No findings were identified.

.6 Risk-Significant High-Radiation Area and Very-High Radiation Area Controls (02.06)

a. Inspection Scope

The inspectors discussed with the radiation protection manager the controls and procedures for high-risk, high-radiation areas and very-high radiation areas. The inspectors discussed methods employed by the licensee to provide stricter control of very-high radiation area access as specified in 10 CFR 20.1602, “Control of Access to Very-High Radiation Areas,” and Regulatory Guide 8.38, “Control of Access to High and Very-High Radiation Areas of Nuclear Plants.” The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become very-high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

The inspectors evaluated licensee controls for very-high radiation areas and areas with the potential to become a very-high radiation areas to ensure that an individual was not able to gain unauthorized access to the very high radiation areas.

b. Findings

No findings were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors observed radiation worker performance with respect to stated radiation protection work requirements. The inspectors assessed whether workers were aware of the radiological conditions in their workplace and the RWP controls/limits in place, and whether their performance reflected the level of radiological hazards present.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the radiation protection manager any problems with the corrective actions planned or taken.

b. Findings

No findings were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to all RWP requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the RWP controls/limits, and whether their performance was consistent with their training and qualifications with respect to the radiological hazards and work activities.

The inspectors reviewed radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

The inspectors evaluated whether problems associated with radiation monitoring and exposure control were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP. The inspectors assessed the appropriateness of the corrective actions for a selected sample of problems documented by the licensee that involve radiation monitoring and exposure controls. The inspectors assessed the licensee's process for applying operating experience to their plant.

b. Findings

No findings were identified.

2RS2 Occupational As-Low-As-Reasonably-Achievable Planning and Controls (71124.02)

The inspection activities supplement those documented in IR 05000255/2014002, and constitute one complete sample as defined in IP 71124.02-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed pertinent information regarding plant collective exposure history, current exposure trends, and ongoing or planned activities in order to assess current performance and exposure challenges. The inspectors reviewed the plant's 3-year rolling average collective exposure.

The inspectors reviewed the site-specific trends in collective exposures and source term measurements.

The inspectors reviewed site-specific procedures associated with maintaining occupational exposures as-low-as-reasonably-achievable, which included a review of processes used to estimate and track exposures from specific work activities.

b. Findings

No findings were identified.

.2 Source Term Reduction and Control (02.04)

a. Inspection Scope

The inspectors used licensee records to determine the historical trends and current status of significant tracked plant source terms known to contribute to elevated facility aggregate exposure. The inspectors assessed whether the licensee had made allowances or developed contingency plans for expected changes in the source term as the result of changes in plant fuel performance issues or changes in plant primary chemistry.

b. Findings

No findings were identified.

2RS4 Occupational Dose Assessment (71124.04)

The inspection activities supplement those documented in IRs 05000255/2014004; 05000255/2015011; and constitute one complete sample as defined in IP 71124.04-05.

.1 Special Dosimetric Situations (02.04)

Assigning Dose of Record

a. Inspection Scope

For the special dosimetric situations reviewed in this section, the inspectors assessed how the licensee assigns dose of record for total effective dose equivalent, shallow dose equivalent, and lens dose equivalent. This included an assessment of external and internal monitoring results, supplementary information on Individual exposures (e.g., radiation incident investigation reports and skin contamination reports), and radiation surveys and/or air monitoring results when dosimetry was based on these techniques.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index—Emergency AC Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System (MS06) performance indicator (PI) for the period from the fourth quarter 2014 through the third quarter 2015. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC Integrated Inspection Reports for the period of October 1, 2014, through September 30, 2015, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI emergency AC power system sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index—Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Cooling Water Systems (MS10) PI for the period from the fourth quarter 2014 through the third quarter 2015. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated Inspection Reports for the period of October 1, 2014, through September 30, 2015, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one MSPI cooling water systems sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Items Entered into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate, timeliness was commensurate with the safety significance, evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrences reviews were proper and adequate, and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

These routine reviews for the identification and resolution of problems did not constitute any additional inspection samples. Instead, by procedure they were considered an integral part of the inspections performed during the quarter and documented in Section 1 of this report.

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for follow-up, the inspectors performed a daily screening of items entered into the licensee's CAP. This review was accomplished through inspection of the station's daily condition report packages.

These daily reviews were performed by procedure as part of the inspectors' daily plant status monitoring activities and, as such, did not constitute any separate inspection samples.

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of July 2015 through December 2015, although some examples expanded beyond those dates where the scope of the trend warranted.

The review also included issues documented outside the normal CAP in major equipment problem lists, repetitive and/or rework maintenance lists, departmental problem/challenges lists, system health reports, quality assurance audit/surveillance reports, self-assessment reports, and Maintenance Rule assessments. The inspectors compared and contrasted their results with the results contained in the licensee's CAP trending reports. Corrective actions associated with a sample of the issues identified in the licensee's trending reports were reviewed for adequacy.

This review constituted one semi-annual trend inspection sample as defined in IP 71152–05.

b. Findings and Observations

The inspectors performed a review of conditions identified in the CAP as potential rework issues to determine if there existed a more significant safety issue. Through the inspectors' review of condition reports written for equipment deficiencies found during the RFO; it was identified that a number of items were categorized as requiring rework evaluations. The inspectors assessed the licensee's identification and screening of equipment issues that could potentially be rework; the evaluation of the cause of rework; the corrective actions taken to prevent the cause of the rework issue from occurring again; and reviewed trend data from the rework evaluations to determine potential common themes. The inspectors also attended the licensee's Station Rework Review Board (SRRB) meeting and reviewed past meeting minutes.

The purpose of evaluating issues for rework is to improve overall plant safety, equipment reliability, and reduce cost/resource use associated with re-performing maintenance on important plant equipment. This is done through evaluation of the issues to identify programmatic or human performance deficiencies that can be corrected to prevent trends and/or significant plant events from occurring. There were 57 condition reports written related to equipment deficiencies identified during or after the RFO that were evaluated for being rework. Of those, 33 were determined to be classified as rework and those evaluations were approved by the SRRB. Most were classified as low level events, having no adverse consequences on plant operation, which is typical from reviewing the other quarterly trend information. About one-third of the rework issues were skill-based (human performance) errors, with the rest being distributed across other areas such as design deficiencies, procedure/instruction adequacy, and vendor deficiencies, to name a few. Also, about one-third of the issues were coded to vendor/supplemental employee work activities. There historically have been trends for outage-related rework issues pertaining to the use of vendors, vendor errors, or supplemental workers. The licensee has written corrective actions to enhance their vendor/supplemental worker oversight and review program. Actions have also been written to enhance and correct the areas in which the rework was grouped, such as revising procedures were errors were identified and correcting parts records or purchase orders when the wrong part was attained. No significant adverse trends were identified and actions/enhancements to address commonalities have been entered into the CAP. The SRRB will roll-up all rework data from calendar year 2015 to potentially identify trends and compare that data to previous years.

.4 Annual Follow-up of Selected Issues: 1-2 Diesel Generator Failure to Start

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors reviewed a corrective action item documenting the failure of 1-2 DG to start during performance of post-maintenance testing on March 18, 2015. The licensee identified that the 1-2 DG did not start due to a failure of the air start motor, ASM-2A. The licensee replaced ASM-2A on March 19, 2015, and post-maintenance testing was completed satisfactorily. Additionally, the licensee performed an equipment apparent cause evaluation (EACE) and sent the failed air start motor to a vendor for failure analysis. The failure analysis identified that the direct cause of the failure of ASM-2A was a sheared stop nut pin in the starter drive assembly. Destructive analysis was performed on the motor and it was determined that the apparent cause of the failure of the stop nut pin was due to inadequate material used in manufacturing the pin. Specifically, the stop nut pin was found to be made of unhardened steel. The characteristics consistent with this material do not have a high enough shear strength for the rated output torque of the air start motor. Testing of a stop nut pin from a similar air start motor identified that the pin had characteristics associated with hardened steel, which is suitable for the air start motor's rated output torque.

The inspectors reviewed the EACE and failure analysis report for identification of the apparent cause. Additionally, the inspectors assessed the consideration of the extent of condition and the evaluation of potential common cause operability issues due to this failure mode. The licensee performed an extent of condition review and operability evaluations on the 1-1 DG ASMs and ASM-2B on the 1-2 DG to determine if this same issue existed on the other currently installed motors. The inspectors also reviewed the licensee's evaluation of operating experience for applicability to this issue. The licensee did not identify any similar operating experience and determined that no similar failures in the industry had occurred. The ASMs installed on the 1-1 DG and ASM-2B were determined to be operable. The licensee also has a corrective action planned to evaluate this issue for applicability to 10 CFR 21, "Reporting of Defects and Noncompliance." Documents reviewed are listed in the Attachment to this report.

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152–05.

b. Findings

No findings were identified.

4OA3 Follow-Up of Events and Notices of Enforcement Discretion (71153)

.1 (Closed) Licensee Event Report 255/2015-001-00: Automatic Reactor Trip Results from a Turbine Trip Initiated from the Digital Electro-Hydraulic Control System

a. Inspection Scope

On September 16, 2015, at approximately 0117, an anomaly within the DEH control system initiated a turbine trip which actuated the reactor protection system to automatically trip the reactor. The direct cause was an actuation of the DEH controller "loss of power" turbine trip logic. Troubleshooting determined the cause of this actuation to be from a combination of a failed power supply module on a circuit board within the DEH overspeed protection control (OPC) unit and either a loss of power from the OPC distributed processing units (DPUs) to the main system data highway or a loss of communication between the primary and backup OPC DPUs. This combination of issues signaled no overspeed protection for the turbine and actuated the trip logic. Prior to restarting the reactor following 1R24, the licensee replaced the failed circuit board and implemented a modification to remove the OPC "loss of power" and "loss of communications" turbine trip logics based on operating experience from other plants that use the same model DEH control system. Future corrective actions will include replacing circuit boards on other susceptible parts of this system in the next two RFOs. These actions are reasonable to prevent recurrence combined with the elimination of the trip logic from the modification that was implemented. The timeframe is acceptable based on the DEH system being single-failure proof and site operating experience of the circuit boards. Documents reviewed are listed in the Attachment to this report. This LER is closed.

This event follow-up review constituted one sample as defined in IP 71153–05.

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2015001–03, Turbine-Driven Auxiliary Feedwater Pump Trip During Surveillance Testing

a. Inspection Scope

The inspectors completed a review of URI 05000255/2015001-03, Turbine-Driven AFW Pump Trip During Surveillance Testing. On November 14, 2014, during performance of the Auxiliary Feedwater (AFW) system 18 month surveillance test procedure, the turbine-driven Auxiliary Feedwater pump (P-8B) tripped on overspeed approximately three minutes after the pump was started. The licensee conducted an equipment apparent cause evaluation (EACE) that identified possible causes and additional testing and data collection occurred in March 2015 to confirm or refute those potential causes. The inspectors reviewed the data and conclusions from the March testing, other subsequent pump testing and system monitoring activities, and corrective actions from the EACE to evaluate the potential causes.

The direct cause of the turbine-driven AFW (TDAFW) pump trip was that the trip valve shut due to actuation of the overspeed trip device. The apparent cause, as stated in the EACE, "concluded that no single probable cause in and of itself caused the trip…several system and program anomalies could not be ruled out and as a result it is concluded that the apparent cause is a combination of multiple probable causes seen." The EACE was revised to indicate five possible causes:

(1) Condensate present in the steam supply can slow down or speed up the turbine while in operation. Through inspections of the steam traps and review of the system during the cause evaluation, it was determined that the moisture removal system was not fully effective and the preventive maintenance program for the safety-related steam traps was not effectively or consistently implemented. The inconsistent equipment classifications and ineffective preventive maintenance strategy for safety-related steam traps is considered a performance deficiency and is documented as a licensee-identified violation in section 4OA7.

(2) Margin between the as-found operating speed and the overspeed setpoint was less than expected. Real-time monitoring of turbine speed, steam supply pressure, and governor valve operation was conducted multiple times during subsequent pump test runs since the event. Any anomalies were documented in the test procedure and dispositioned through the corrective action program. The overspeed trip setpoint was revised based on the vendor's recommendation for the operating conditions of this particular pump.

(3) Inherent design conditions of the steam supply and control systems could lead to condensate buildup which could change the turbine speed or cause pressure oscillations at the steam supply inlet. As mentioned in number 2 above, real-time monitoring of the system has been implemented during test runs and any issues are documented in the corrective action program. The steam supply piping from the 'A' steam generator was mapped and determined to have a section of negative slope which had the potential for condensate buildup in the piping upstream of the turbine steam supply valve. The licensee is currently evaluating options for enhancing the draining capability of this line. However, these actions are considered enhancements to the system and are not likely causes of the trip.

(4) Testing the operation of the turbine trip valve while conducting pre-test checks provides an increased frequency for error in properly resetting the trip mechanism. The auxiliary feedwater pumps test procedures were reviewed to identify instances where the overspeed trip valve was tested prior to running the pump. The quarterly technical specification surveillance was revised to remove this testing and the overspeed trip valve will only be tested on a refueling cycle frequency going forward.

(5) Other owners of Elliott turbines have raked knife edges and reset the lever/latch plate assembly differently. The licensee has completed an engineering change

to install a raked latch plate and knife edge on the overspeed trip assembly. This will be completed in February 2016 during the next pump maintenance window.

Based on the inspectors' review of the licensee's operational data collection and evaluation, and subsequent successful testing of the pump, this unresolved item is being closed. Documents reviewed are listed in the attachment.

b. Findings

There was one licensee-identified violation related to this URI that is documented in section 4OA7. This URI is closed.

.2 (Closed) NRC Temporary Instruction 2515/190, Inspection of the Licensee's Proposed Interim Actions As A Result of the Near-Term Task Force Recommendation 2.1 Flooding Reevaluation

a. Inspection Scope

Inspectors verified that licensee's interim actions will perform their intended function for flooding mitigation.

The inspectors independently verified that the licensee's proposed interim actions would perform their intended function for flooding mitigation.

- Visual inspection of the flood protection feature was performed if the flood protection feature was relevant. External visual inspection for indications of degradation that would prevent its credited function from being performed was performed.

- Reasonable simulation, if applicable to the site.

- Flood protection feature functionality was determined using either visual observation or by review of other documents.

The inspectors verified that issues identified were entered into the licensee's CAP.

These activities constituted the completion of Temporary Instruction (TI) 2515/190, Inspection of the Proposed Interim Actions Associated with Near-Term Task Force Recommendation 2.1 Flooding Hazard Evaluations.

b. Findings and Observations

The licensee identified that potential interim actions required for a Beyond Design Basis Local Intense Precipitation event were not identified in PNP 2015-018, "Required Response 2 for Near-Term Task Force Recommendation 2.1: Flooding – Hazard Re-Evaluation Report." The report discussed a water ingress path to the North Penetration Room (Door #106) that eventually leads, through an elevated pipeway, into the safety related 1D Switchgear Room. The licensee's report stated that flooding during rainfall had never been recorded in the area and the "entire room would have to flood before reaching the pipeway." On November 19, 2015, the licensee identified an additional pipeway inside the North Penetration Room and that the pipeway from the electrical penetration room to the 1D switchgear room was not elevated. The licensee entered this issue into their CAP, evaluated the potential water ingress path, and determined that additional interim actions were needed. The licensee identified an additional interim action to place sandbags across the doorway to the North Penetration Room in the event that heavy rain is predicted. In addition to this interim action, the inspectors reviewed other interim actions and procedures to implement these actions. Materials for performing these activities were onsite and licensee staff was familiar with their use.

No findings were identified. This TI is closed.

.3 Safety-Conscious Work Environment Issue of Concern Follow-up (IP 93100)

a. Inspection Scope

The inspectors reviewed licensee actions to assess if they addressed previously identified work environmental conditions that potentially impacted licensee employee's willingness and ability to raise issues impacting nuclear safety. Specifically the inspectors reviewed documents and conducted focus group and individual interviews to determine if:

- indications of a chilled work environment existed;
- employees were reluctant to raise safety or regulatory issues; and
- employees were being discouraged from raising safety or regulatory issues.

The inspectors also reviewed the results of a safety culture assessment conducted for the licensee by an independent contractor. Documents reviewed are listed in the Attachment to this report.

b. Findings

No findings were identified. The inspectors' interviews and focus groups and a recent licensee-initiated safety culture survey indicated that most individuals felt free to raise safety or regulatory issues without fear of retaliation. The inspectors did not identify indications of a chilled work environment at the plant. Additionally, the safety culture assessment survey indicated that the culture of the station had improved since the previous assessment conducted approximately 3 years ago.

c. Observations

The inspectors conducted four focus groups with Security Officers and one group with Mechanical Maintenance personnel for a total of 40 individuals. The inspectors also conducted nine one-on-one interviews with Security Officers in the field. In addition, the inspectors interviewed some of the station management team, including the Employee Concerns Program (ECP) manager, and conducted a review of the ECP files from 2015.

Overall, the responses from the interviews were that personnel were encouraged and willing to raise nuclear safety concerns and did not feel that they would suffer retaliation for raising issues. Some of the individuals expressed a reluctance to write condition reports because they did not receive feedback on the issues and therefore did not feel like their concerns were effectively resolved. However, these individuals stated that they would raise issues to their first line supervisors and resolve them through that avenue. Through the interviews and focus groups, it was highlighted to the inspectors that there continues to be issues with trust of the ECP program. Plant personnel knew who the ECP manager was and where the ECP office was located, but they did not feel that the ECP was always confidential.

The Security organization indicated that they were fully staffed and that overtime was at an all-time low. This was an improvement over previous responses obtained in focus groups and interviews, where concerns were expressed on the shortage of staff and resources and the increased use of overtime to ensure full crew compliments. The interviews continued to reveal challenges with regard to communication, particularly from the senior management team down to the line organization. This has been a continuing area for improvement within the security department and the entire Palisades organization. For example, there have been numerous opportunities for job placement outside of the Security Organization, including some opportunities for additional training within the department. However, when asked if the officers were aware of these openings, they indicated that applying for those positions was not of benefit because they did not believe they would be hired or given the chance to try something different. So, while these opportunities may have been available to the staff, the conditions for pursuing them were not clearly communicated in a manner that encouraged individuals to try them.

The inspectors also reviewed actions in the Confirmatory Order action list that were implemented by the security department. While the majority of items from the action list were being effectively implemented, including improvement in the safety culture, some items in the action list were not being effectively utilized. Specifically, the “Questions and Concerns” notebook, in which officers are able to write down questions or concerns they may have and acts as a log for keeping answers to those questions, was not being effectively utilized. When asked about the notebook, the officers indicated that it was available for a couple of weeks and then disappeared. The inspectors located and reviewed the notebook and noted that the last entry was from May 2015. Additionally, the Security Ombudsman program is not apparent to the officers within the department, which was another tool that was intended to be utilized to improve communications. The inspectors communicated the observations of the implementation of the items in the action list to the licensee management team.

There have been a series of safety culture surveys conducted in 2015, however, at the time of the inspection, none of the results had been communicated to the line organization. Station personnel had received information regarding the response rates for the surveys for each organization, but the actual survey data and results had not been communicated. This could lead to the potential for the workforce to become apathetic about participating in the surveys at all. In addition, some members of the licensee’s staff said they felt that many of the questions, as worded in the surveys, did not pertain specifically to their jobs or roles within the organization.

The licensee commissioned an independent third-party safety culture assessment survey which was conducted in 2015. That assessment had an overall participation response rate of 76 percent and compared results with a previous equivalent assessment survey in 2012. The specific primary areas/dimensions measured were Nuclear Safety Culture, General Cultural and Work Environment, and Leadership, Management, and Supervisory Behaviors and Practices. Three sub- dimensions measured were Employee Concerns Program, Nuclear Safety Values, Behaviors, Practices, and Safety Conscious Work Environment. The 2015 assessment survey found “a notable improvement of +6.4 %” in cultural scores since the 2012 assessment. Five of six areas/dimensions measured were identified as an “Area of Strength,” the sixth area (Employee Concerns Program) was considered an “Area of Competency.” Inspectors did not identify any items from the interviews that were at significant variance with the assessment survey results.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 12, 2016, the inspectors presented the inspection results to Mr. A. Vitale, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the Radiological Hazard Assessment and Exposure Controls inspection during 1R24 were presented to Mr. A. Vitale, Site Vice President, and other members of the licensee staff on October 2, 2015;

- The inspection results of the biennial In-Service Inspection during 1R24 were presented to Mr. A. Vitale, Site Vice President, and other members of the licensee staff on October 7, 2015;

- The results of the inspectors’ review of URI 05000255/2014008-11, “Classification of CCW Piping and Components Inside the Reactor Containment Building,” was presented to Mr. A. Vitale; Site Vice President and other members of the licensee staff on October 29, 2015;

- On November 4, 2015, the inspectors debriefed the preliminary results of the Safety Conscious Work Environment follow-up inspection to the Site Vice President, Mr. A. Vitale, and other members of the licensee staff. On January 7, 2016, the inspectors presented inspection results to the Site Vice President, Mr. A. Vitale, and other members of the licensee staff. The licensee acknowledged the issues presented;

- The annual review of Emergency Action Level and Emergency Plan changes was presented to Mr. D. Malone, Emergency Preparedness Manager, and other members of licensee staff via telephone on November 24, 2015;

- The inspection results of the Triennial Review of Heat Sink Performance were presented to Mr. A. Williams, General Manager, for Plant Operations, and other members of the licensee staff on November 20, 2015, and

- The inspection results of the CDBI activities related to the closure of URI 05000255/2014008-11 were presented to Mr. Vitale, and other members of the licensee’s staff on October 29, 2015.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

4OA7 Licensee-Identified Violations

The following violation of very low 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 an NCV.

- Title 10 CFR 50.65(a)(1), requires, in part, that the holders of an operating license shall monitor the performance or condition of structures, systems, and components (SSCs), against licensee-established goals, in a manner sufficient to provide reasonable assurance that these SSCs, as defined in 10 CFR 50.65(b), are capable of fulfilling their intended functions. Title10 CFR 50.65(a)(2) states that monitoring as specified in 50.65(a)(1) is not required, where it has been demonstrated that the performance or condition of a SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function. Contrary to the above, as identified after the November 14, 2014, TDAFW pump trip, the licensee failed to demonstrate the performance or condition of the safety-related auxiliary feedwater system steam traps had been effectively controlled through the performance of appropriate preventive maintenance. Specifically, some of the safety-related steam traps, one relief valve, and one check valve associated with the steam supply piping of the turbine-driven AFW system were inappropriately classified in the maintenance rule program, resulting in inadequate and/or untimely maintenance being performed on these components, which probably contributed to the overspeed trip event. The licensee found 3 steam traps and one relief valve classified as non-critical components that were reclassified as high critical components and one steam trap and one check valve classified as run-to-failure components that were reclassified as high critical components. Some of these components also had no preventive maintenance (PM) strategies or ones that were not the correct frequency based on the component classification. The licensee identified this issue while conducting the equipment apparent cause evaluation for the overspeed trip event and documented actions to correct the issue in CR-PLP-2014-5477. The licensee performed inspections of all the steam traps required for the TDAFW pump operation and identified some issues with steam cutting, foreign material exclusion in the traps, and incomplete seat contact. These issues were corrected and PM changes have been made for all the system components mentioned above.

  The inspectors determined that the inconsistent equipment classifications and ineffective preventive maintenance strategy for the safety-related steam traps in the turbine-driven auxiliary feedwater system is considered a performance deficiency. The performance deficiency was more than minor because it was associated with the Equipment Performance attribute of the Mitigating Systems cornerstone and adversely impacted the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events. Specifically, the licensee identified that the degraded condition of the moisture removal system could have led to excess condensate being present in the steam supply line which had the potential to adversely affect the operation of the turbine for the TDAFW pump, contributing to the overspeed trip event. The inspectors screened the issue using IMC 0609, Appendix A, “The SDP for Findings at Power,” Exhibit 2, “Mitigating Systems Screening Questions,” and answered “Yes” to the question of “does this finding represent a loss of system and/or function?” This trip of the TDAFW pump on overspeed was evaluated as a failure that impacted the ability of the AFW system to provide the specific function, which could only be accomplished by this train, of decay heat removal via steaming of the ‘A’ Steam Generator. The turbine-driven AFW pump was also determined to not be in a condition to meet performance requirements defined by the probabilistic risk assessment success criteria, which for AFW is a 24 hour mission time. Therefore, the issue was screened further in a detailed risk evaluation.

A Region III Senior Reactor Analyst performed a detailed risk evaluation using the NRC’s Standardized Plant Analysis Risk Model for Palisades, Revision 8.20. The SRA assumed the turbine driven AFW pump was unavailable to perform its’ function for a period of 3 days because the pump was successfully tested and returned to service on November 16, 2014. Given the short exposure period, the calculated delta core delta frequency was less than 1.0E-7/yr. As a result of the low calculated delta core delta frequency, no additional analysis of external event risk contribution or large early release risk contribution was necessary. The dominant core damage sequence was a station blackout followed by the failure of the turbine driven AFW pump and the failure to recover onsite or offsite power. Therefore, the finding screened as very low safety significance (Green).

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

A. Vitale, Site Vice President
B. Baker, Operations Manager – Shift
J. Borah, Engineering Manager, Systems and Components
E. Chatfield, Employee Concerns Coordinator
R. Craven, Production Manager
T. Davis, Licensing Specialist
B. Dotson, Licensing Specialist
J. Erickson, Regulatory Assurance
T. Mulford, Operations Manager
D. Nestle, Radiation Protection Manager
T. Fouty, Engineering Supervisor
O. Gustafson, Director of Regulatory and Performance Improvement
J. Hardy, Regulatory Assurance Manager
J. Haumersen, Site Projects and Maintenance Services Manager
G. Heisterman, Maintenance Manager
M. Lee, Operations Manager - Support
M. Liska, System Engineer
D. Lucy, Outage Manager
D. Malone, Emergency Planning Manager
D. MacMaster, Engineering Supervisor
W. Nelson, Training Manager
D. Nestle, Radiation Protection Manager
K. O'Connor, Engineering Manager, Design and Programs
C. Plachta, Nuclear Independent Oversight Manager
P. Russell, Site Engineering Director
M. Schultheis, Performance Improvement Manager
M. Soja, Chemistry Manager
K. Strickland, Environmental Specialist
J. Tharp, Security Manager
A. Williams, General Manager Plant Operations

U.S. Nuclear Regulatory Commission

E. Duncan, Chief, Branch 3

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000255/2015004-01	NCV	Inadequate Dye Penetrant Examination of Pipe Lug Welds (Section 1R08.1)
05000255/2015004-02	NCV	Failure to Identify Components Required to be Covered by the Quality Assurance Program (Section 1R21.b.(1))
05000255/2015004-03	SL IV	Failure to Provide Bases to Determine Changes Did Not Involve Unreviewed Safety Questions (Section 1R21.b.(2))
05000255/2015004-04	SL IV	Failure to Perform a Required 50.59 Evaluation for Declassification of the CVCS (Section 1R21.b.(3))

Closed

05000255/2015004-01	NCV	Inadequate Dye Penetrant Examination of Pipe Lug Welds (Section 1R08.1)
05000255/2014008-11	URI	Classification of CCW Piping and Components Inside the Reactor Containment Building (Section 1R21.a)
05000255/2015004-02	NCV	Failure to Identify Components Required to be Covered by the Quality Assurance Program (Section 1R21.b.(1))
05000255/2015004-03	SL IV	Failure to Provide Bases to Determine Changes Did Not Involve Unreviewed Safety Questions (Section 1R21.b.(2))
05000255/2015004-04	SL IV	Failure to Perform a Required 50.59 Evaluation for Declassification of the CVCS (Section 1R21.b.(3))
2515/190	TI	Inspection of the Licensee's Proposed Interim Actions As A Result of the Near-Term Task Force Recommendation 2.1 Flooding Reevaluation (4OA5.2)
05000255/2015001–03	URI	Turbine-Driven Auxiliary Feedwater Pump Trip During Surveillance Testing (4OA5.1)

Discussed

05000255/2014008-11	URI	Classification of CCW Piping and Components Inside the Reactor Containment Building (Section 1R21.a)

LIST OF DOCUMENTS REVIEWED

The following is a partial list of documents reviewed during the inspection. Inclusion on this list does not imply that the NRC inspector reviewed the documents in their entirety, but rather that selected sections or 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.

1R01 Adverse Weather

- AOP-38; "Acts Of Nature;" Revision 4
- CR-PLP-2015-05628; Based On A National Weather Service High Wind Warning, Entry Into AOP-38, Acts of Nature Is Required; Dated November 11, 2015
- CR-PLP-2015-05646; Blue Tarp Caught in the Razor Wire; Dated November 12, 2015
- EN-FAP-EP-010; "Severe Weather Response;" Revision 2

1R04 Equipment Alignment

- GOP-11; Attachment 4, GCL 11.4, Refueling/Fuel Handling/Shutdown Cooling Ventilation Checklist; Revision 48
- GOP-14; Shutdown Cooling Operations; Revision 49
- M-202; Chemical & Volume Control System, Sheet 1A; Revision 64
- M-202; Chemical & Volume Control System, Sheet 1B; Revision 54
- M-203; Safety Injection, Containment Spray and Shutdown Cooling System, Sheet 2; Revision 27
- M-204; Safety Injection Containment Spray, Sheet 1B; Revision 4
- M-204; Safety Injection, Containment Spray and Shutdown Cooling System, Sheet 1A; Revision 43
- M-218; Heating, Ventilation and Air Conditioning Containment Building, Sheet 2; Revision 61
- M-221; Spent Fuel Cooling System, Sheet 2; Revision 59
- SOP-20; Station Power; Revision 77
- SOP-27; Fuel Pool system; Revision 69
- SOP-2A; Chemical and Volume Control System; Revision 85

1R05 Fire Protection

- Administrative Procedure 4.49; Non Power Operation Fire Risk Management; Revision 0
- Calculation EA-FPP-03-001; Analysis of Combustible Loading at Palisades Nuclear Plant; Revision 3
- CR-PLP-2015-05417; Door 166, Spent Fuel Pool North Stairwell Door was Unsat for Safety Related Fire Door Inspection; Dated November 1, 2015
- CR-PLP-2015-5701; During Performance of FPSP-SO-4 Found That MV-FP723 was Very Difficult to Operate; Dated November 18, 2015
- CR-PLP-2015-5774; Communications Equipment Located in SFP and Ventilator 8A & B Rooms Does Not Appear to Be Addressed; Dated November 20, 2015
- Design Basis Document DBD-7.10; NFPA 805, Fire Protection Program; Revision 0
- Drawing M-216; Sheet 2, Fire Protection System; Revision 71
- EN-DC-127; Control of Hot Work and Ignition Sources; Revision 15
- EN-DC-161; Control of Combustibles; Revision 13
- EN-DC-359; Fire Risk Management During Non-Power Operations for NFPA 805 Plants; Revision 1
- FPIP-4; Fire Protection Systems and Fire Protection Equipment; Revision 36

- GOP-14; Shutdown Cooling Operations; Revision 49
- OL-OLPLP-2014-026; Outage Risk Assessment RO-24; Revision 1
- Palisades Nuclear Plant Fire Hazards Analysis; Revision 7
- Pre-Fire Plan 13B; Charging Pump Rooms / Auxiliary Building - Elevation 590’
- Pre-Fire Plan 13G; Spent Fuel Pool Heat Exchanger Room / Auxiliary Building - Elevation 590’
- Pre-Fire Plan 17; Refueling & Spent Fuel Pool Area – Elevation 649’
- Pre-Fire Plan 24; Auxiliary Feedwater Pumps Room / Turbine Building - Elevation 571’
- WO 52629311-01; FPSP-SO-4 – Fire Suppresion Water System PIV Operation
- WT-WTPLP-2012-0351; Revise Calculation EA-FPP-03-001; “Analysis of Combustible Loading at Palisades Nuclear Plant,” Dated October 22; 2012

1R07 Heat Sink Performance

- 02-07-203.64; North & South Intake Structure Inspection & Cleaning Fall 2013; Dated November 11; 2013
- 02-07-203.66; Service Water Bay Inspection; Dated February 13, 2014
- 02-07-203.70; Inspection and Cleaning of F-4B and F-4C Bays; Dated May 14, 2014
- AOP-21; Abnormal Operating Procedure -EDG 1-2 Malfunctions; Revision 1
- AOP-35; Loss of Service Water; Revision 0
- AOP-38; Acts of Nature; Revision 4
- ARP-7; Auxiliary Systems Scheme EK-11 (C-13); Revision 95
- Basis Document for TS Surveillance Procedures MO-7A-1 and MO-7A-2; Revision 12
- C-221; Intake Structure Plan; Revision 21
- C-222; Intake Structure Sections; Revision 1C
- C-224 SH2; Intake Structure Details; Revision 4
- Calculation EA-D-PAL-93-272F-02; Diesel Generators Lube Oil Cooler and Jacket Water Cooler Performance Analysis; Revision 0
- Calculation EA-EC28106-02; Revision 0 per EC28106 Verification of HTRI Xist Software for LO Cooler; Dated February 27; 2013
- Calculation EA-EC28106-03; Diesel Generator Lube Oil Cooler Tube Plugging; Revision 0
- CCW; DG 1-1 & 1-2 and Fire Water Systems Report; Dated January 22, 2015
- COP-15; Service Water System Chemistry; Revision 22
- CR PLP 2015 05728; NRC Identified Calculations; e.g.; EA-D-PAL-93-272F-02 Assumed Maximum Temperature of Jacket Water Exiting Engine Was 190 degrees Fahrenheit But Procedures; e.g.; AOP-20; Did Not Trip EDG Until 195 Degrees Fahrenheit; initiated November 19, 2015
- Critical Service Water System Health Report; Dated Q2 2015
- Critical Service Water System Health Report; Dated Q1 2015
- CR-PLP-2011-01192; 1-2 EDG lube Oil Temperature Below Normal Band of SOP-22 Attachment 6 with Diesel at Full Load with Operability Determination; Dated March 12; 2011
- CR-PLP-2012-06534; Engineering Change (EC) 26869 and EA-D-PAL-93-272F-02 were Completed in Winter of 2011 to Determine Tube Plugging Allowance for EDG E-22A/B Jacket Water Coolers and E-31A/B Lube Oil Coolers; Dated October 2; 2012
- CR-PLP-2012-07527; Maintenance Rule Component Failure Associated with Ultimate Heat Sink; Dated December 04; 2012
- CR-PLP-2013-01233; The EDG E-31B Lube Oil Cooler calibration Standard Used is the Current Standard and There Are Issues With These Eddy Current Inspections; Dated March 18; 2013
- CR-PLP-2014-04246; Spare EDG Lube Oil Cooler Lacked FME Covers On Two Pipe Openings On The Shell Side Of The Spare Lube Oil Cooler; Initiated August 27, 2014

- CR-PLP-2015-00348; Deep Corrosion Identified In Some Areas Of E-31B Lube Oil Cooler Sealing Gasket With Operability Determination; Initiated January 21, 2015
- CR-PLP-2015-02078; HCM Transition From 1 Onsite HX Expert To Systems Engineering Has Not Been Effectively Implemented; Dated May 19, 2015
- CR-PLP-2015-02462; ECT Inspection of E-31A; 1-1 EDG Lube Oil Cooler Identified 3 tubes Requiring Plugging With Operability Evaluation; Initiated April 8, 2014
- CR-PLP-2015-04504; Service Water Flow To EDG-1 Did Not Meet The Acceptance Criteria; Dated September 30, 2015
- CR-PLP-2015-05064; Investigated Tube Plugging Limit for VHX-1 Using An Outdated Engineering Calculation; Dated October 15, 2015
- CR-PLP-2015-05318; Deficiency Identified During Pre-NRC UHS Assessment; Dated October 26, 2015
- CR-PLP-2015-05520; Justification for not testing in the Master Heat Exchanger Testing Plan (Revision 12) Needs Improvement; Dated November 5, 2015
- CR-PLP-2015-05644; HX With Raw Water Coded as Mild Environment; Dated November 12, 2015
- CR-PLP-2015-05842; NRC Resident Identified Discrepancy In The VHX-1 Inspection Conducted Under WO 52544850-09; Dated November 24, 2015
- DBD-1.02; DBD for Service Water System; Revision 8
- DBD-1.08; DBD Ultimate Heat Sink; Revision 4
- Drawing M-208; P&ID for Service Water System; sheet 1A; Revision 65
- Drawing M-214; P&ID for Lube Oil; Fuel Oil and Diesel Generator Systems; sheet 1; Revision 79
- DWO-1; Operator's Daily/Weekly Items Modes 1,2,3; and 4; Revision 107
- EA-C-PAL-99-1209B-01; Generation of Flow Rate Acceptance Criteria for Technical Specifications Surveillance Test RO-216
- EA-C-PAL-99-1209B-01; Generation of Flow Rate Acceptance Criteria for Technical Specifications Surveillance Test RO-216; Revision 3
- EA-D-PAL-93-272F-02; Diesel Generators Lube Oil Cooler and Jacket Water Cooler Performance Analysis; Revision 0
- EAR 98-0512; Establish 85 F as Design Basis Service Water Temperature Limit; Revision 0
- EC 37737; Approval of Spare Service Water Pump Bowl from Hydroaire; Revision 0
- EC 40156; Acceptance of Refurbished Discharge Column Assembly; Revision 0
- EC 53879; Line Shaft Coupling Material Change for P-5; Revision 0
- Emergency Diesel Generator System Health; Q1-2015
- Emergency Diesel Generator System Health; Q2-2015
- Emergency Diesel Generator System Health Report; Dated Q1 2015
- Emergency Diesel Generator System Health Report; Dated Q2 2015
- EN-DC-150; Condition Monitoring of Maintenance Rule Structure; Revision 8
- EN-DC-184; NRC Generic Letter 89-13 Service Water Program; Revision 3
- EN-DC-316; Heat Exchanger Performance and Condition Monitoring; Revision 7
- Final ECT Report for E31B LO Cooler and E22B JWHx; Dated March 2013
- Final Eddy Current Inspection Report of the Emergency Diesel Generator 1-2 K6B Lube Oil Cooler 31B Jacket Water Cooler 22B; Dated March 20; 2013
- Final Eddy Current Inspection Report of the Emergency Diesel Generator 1-2 K6B Lube Oil Cooler 31B; Dated January 2015
- GE Locomotive Design Basis Documentation 1.3 Heat Exchangers; C843 1197
- Heat Exchanger Life Cycle Management Plan; Revision 6
- M-124; Heating & Ventilation Reactor Containment Building Coolers – Unit V-1 & Unit V-2; Sheet 1; Revision 7
- M-208 SH1; Piping & Instrument Diagram Non-Critical Service Water System; Revision 105

- M-208 SH1A; Piping & Instrument Diagram Service Water System; Revision 65
- M-208 SH1B; Piping and Instrument Diagram Service Water System; Revision 39
- M-213; Piping and Instrument Diagram Service Water; Screen Structure and Chlorinator; Revision 95
- MO-7A-1 and 2; Palisades Nuclear Plant TS Surveillance Procedure Basis Document; Revision 12
- MO-7A-2; EDG 1-2 Technical Specification Surveillance Procedure; Revision 87; With Full Loading Completed on December 14, 2015
- MO-7A-2; EDG 1-2 TS Surveillance Procedure; Revision 67; With Peak Attachment 6 Loading Completed on January 23; 2009; by WO 51793891
- MO-7A-2; EDG 1-2 TS Surveillance Procedure; Revision 72; With Peak Attachment 6 Loading Completed on March 13; 2013
- MO-7A-2; EDG 1-2 TS Surveillance Procedure; Revision 87; With Full Loading Completed on December 15, 2014
- Palisades EDG Lube Oil Cooler Specification Sheet Reference No. DE-35226; Dated January 21; 1969
- Palisades Nuclear Plant Master Heat Exchanger Testing Plan; Revision 12
- PLP-RPT-12-00113; Raw Water Corrosion Program Report Cycle 22 and 2012 Refueling Outage; Revision 0
- PM Optimization Worksheet; Replace Heat Exchanger Service Water Inlet And Outlet Expansion Joints; Dated December 23; 2003
- Procedure No ARP-20B; Diesel Generator 1-2 Scheme EK-30; Revision 8
- Procedure No COP-22; Diesel Generator Cooling Water Chemistry; Revision 23
- RO-144; Comprehensive Pump Test Procedure Service Water Pumps P-7A; P7B and P-7C
- RU14-021; Internal ROV Inspection of the Raw Water Intake Pipeline Servicing Palisades Nuclear Power Plant; Dated November 8; 2007
- SEP-HX-PLP-001; Attachment 1; Requirements For Generic Letter 89-13 Heat Exchangers; Revision 2
- SOP-15; Service Water System; Revision 60
- Vendor Manual for Fairbanks Morse ALCO Power Inc. Installation; Operation and Maintenance American Standard Type CP and CPR Heat Exchangers LO Cooler and Jacket Water Heat Exchanger; Dated November 1963
- VEN-M-101 SH 2810; Service Water To Engineering Safeguards; Revision 8
- VEN-M-101 SH 2811; Service Water To Engineering Safeguards; Revision 7
- VEN-M101 SH5011; Warm Water Line To Intake Structure; Revision 3
- VEN-M-60A; Containment Air Coolers VHX-1; Cooling Coil Tube Map; Sheet 7(1); Revision 1
- VTD-0207-0135; Fairbanks Morse; ALCO Power Inc. Installation; Operation; and Maintenance Instructions for American Standard Type CP and CPR Heat Exchangers; Dated November 1963
- WO 277999 03; Traveling Screen; F-4B Has a broken Shear Pin
- WO 51652252; K-6B; XJ-0803; Replace Expansion Joints
- WO 52204182; K-6B; Inspect and Perform Pressure Test On Aftercooler
- WO 52262018 01; Misc SWS Basin Level Instruments
- WO 52365980; RT-71G-3 – SEC 5.4; “Outside Containment” SWS Class 2/3 Test
- WO 52419832 01; Misc SWS Basin Level Instrument Calibration
- WO 52435748 01; Divers to Clean/Inspect Service Water Pump Bay
- WO 52485819; K-6A; 24M (1Cycle) PM of Aftercooler & Heat Exchangers
- WO 52522380; K-6B; 24M (1Cycle) PM of Aftercooler & Heat Exchangers
- WO 52537659; RO-216 – Service Water Flow Verification
- WO 52544850; VHX-1; Containment Air Cooler Inspect and Test
- WO 52560090; V-1A & B; Containment Air Cooler Filter Post-Maintenance Test

- WO 52563836 01; Divers to Clean Service Water Pump Bay
- WO 52592110 01; F-4C; Insp/Lube CPLG & Drive Sprocket PM
- WO 52592389 01; Diver Inspection / Cleaning of Intake Bay
- WO 52594558 01; Divers Inspection / Cleaning of Intake Bay
- WO 52594558-01; Diver Inspection / Cleaning East of Intake Bay
- WO-PLP-52369505; Performed PM To Inspect And Clean And Repair E31B LO Cooler
- WO-PLP-52522380; Performed PM To Inspect And Clean And Repair E31B LO Cooler

<u>1R08 Inservice Inspection Activities</u>

- CR-PLP-2014- 01135; BMV Relevant Indication; Dated February 7, 2014
- CR-PLP-2014-00706; Dated January 27, 2014
- CR-PLP-2014-00713; Dated January 27, 2014
- CR-PLP-2014-01073; Ultrasonic Examination Revealed Two Axial Flaws; Dated February 5, 2014
- CR-PLP-2014-02127; Boric Acid on MV-PC1038A; Dated March 14, 2014
- CR-PLP-2014-0343; Boric Acid on PCS P-50A; Dated January 29, 2014
- CR-PLP-2014-03995; Dated August 5, 2014
- CR-PLP-2014-05569; CRD Shock Mount Assemblies Not in ISI Program; Dated November 20, 2014
- CR-PLP-2015- 04636; BMV Relevant Indication; Dated October 3, 2015
- CR-PLP-2015-00592; CRD and ICI Welds Not in Active ISI Database; Dated February 5, 2015
- CR-PLP-2015-04191; NRC Observations During Liquid Penetrant Exam; Dated September 28, 2015
- CR-PLP-2015-04414; RT Procedure Non-Compliance; Dated September 28, 2015
- Drawing 424558; Repair Spray Nozzle Loop 2A Palisades; Revision 2
- Drawing M00001CB1-0819; CRDM Shock Mount Assembly; Revision 0
- EPRI PDQS; 54-ISI-604, Revision 11; Dated August 19, 2014
- Evaluation 14-PAL-0032; PCS P-50B, Primary Coolant Pump Seal Leak; Dated January 19, 2014
- Evaluation 14-PAL-0099; MV-PC600, Downstream Cap Leak; Dated March 14, 2014
- Evaluation 14-PAL-0187; CV-1057, Pressurizer Spray Valve Packing Leak; Dated June 21, 2014
- Nondestructive Examiner Certification Review and Approval Forms; Dated September 17, 2015
- PAL1R24 ETSS # 1 – Bobbin; Revision 1
- PAL1R24 ETSS # 2 – 3-Coil RPC; Revision 2
- PAL1R24 ETSS # 3 – 1-Coil RPC; Revision 1
- PQR-08-08-TS-001; Dated August 1, 2011
- PQR-08-08-TS-002; Dated January 20, 2010
- PQR-08-43-T-001; Dated November 4; 2011
- PQR-A843256-52; Dated March 30; 1992
- Procedure 03-127284; Field Procedure for Remote Rolled Plugging Utilizing the LAN SAP Box; Revision 20
- Procedure 03-9175728; Palisades Unit 1 Steam Generator Eddy Current Analysis Guidelines; Revision 2
- Procedure 54-ISI-400-021; Multi-Frequency Eddy Current Examination of Tubing; Revision 21
- Procedure 54-ISI-400-021; Multi-Frequency Eddy Current Examination of Tubing; Revision 21
- Procedure 54-ISI-460-004; Multi-Frequency Eddy Current Examination of Nozzle Welds and Regions; Revision 4

- Procedure 54-ISI-494-000; Multi-Frequency Eddy Current Array Probe Examination of Vent Line and RVLIS Nozzle Bores; Revision 0
- Procedure 54-ISI-604-013; Automated Ultrasonic Examination of Open Tube RPV Closure Head Penetrations; Revision 13
- Procedure CEP-NDE-0641; Liquid Penetrant Examination for ASME Section XI; Revision 7
- Procedure CEP-NDE-0731; Magnetic Particle Examination for ASME Section XI; Revision 4
- Procedure CEP-NDE-0955; Visual Examination of Bare Metal Surfaces; Revision 303
- Procedure EN-DC-319; Boric Acid Corrosion Control Program; Revision 11
- Procedure LMT-10-PAUT-007; Fully Encoded Phased Array Ultrasonic Examination of Dissimilar Metal Piping Welds; Revision 1
- Procedure SEP-BAC-PLP-001; Boric Acid Corrosion Control Program; Revision 2.
- Procedure SEP-SG-PLP-001; PLP Steam Generator Program; Revision 2
- Report 1R21-VT-15-101; RPVCH Surface; Dated October 6, 2015
- Report 1R23-VE-14-004; Ultrasonic Examination PCS-6-PRS-1C1-1; Dated February 11, 2014
- Report 1R23-VT-14-102; RPVCH Surface; Dated February 10, 2014
- Report 1R24-PT-15-003; ESS-12-SIS-1a1-3PL1-4(H739); Dated September 27, 2015
- Report 1R24-VE-15-009; 2 inch Cold Leg Charging Nozzle Weld PCS-30-RCL-1A-11/2; Dated October 5, 2015
- Report 1R24-VE-15-015; Shutdown Cooling Safe-End to Elbow Weld PCS-12-SDC-2H1-2; Dated October 6, 2015
- Report 1R24-VE-15-020; 2 inch Cold Leg Drain Nozzle Weld PCS-30-RCL-2A-5/2; Dated October 4, 2015
- Report 1R24-VE-15-022; 2 inch Hot Leg Drain Nozzle Weld PCS-42-RCL-1H-3/2; Dated October 6, 2015
- Report 210773-PT-WEG-07; Dated February 22, 2014
- Report 51 – 9216023-002; Qualified Eddy Current Techniques for Palisades Outage 1R24; September 2015; Revision 2
- Report 51 - 9243395-000; Steam Generator Degradation Assessment for Palisades 1R24 Inspection; Fall 2015
- Report BOP-RT-14-002; Dated February 24, 2014
- Report BOP-RT-14-003; Dated February 24, 2014
- Report Liquid Penetrant Examination Report; CRDM Stalk; Dated June 21; 2001
- Report Palisades Unit 1; R024; Reactor Head Inspection Report; Dated October 4, 2015
- Report PLP-RPT-14-00014; 1R23 Steam Generator Condition Monitoring Assessment for 2014 Refueling Outage and Final Operational Assessment for Cycle 24; Revision 1
- Report Root Cause Evaluation Report (CR-PLP-2012-05623); Revision 3
- WDS-002-2A-1; PCS-3-PSS-2A1-1-X1; Dated February 28, 2014
- WDS-002-2A1-3; PCS-3-PSS-2A1-3-X3; Dated February 28, 2014
- WPS-08-08-TS-001; Revision A
- WPS-08-43-T-001; Revision 5

1R11 Licensed Operator Requalification Program

- AOP-25; Loss of Refueling Water Accident; Revision 0
- AOP-26; Loss Of Spent Fuel Pool Cooling; Revision 2
- AOP-26; Loss of Spent Fuel Pool Cooling; Revision 2
- AOP-30; Loss of Shutdown Cooling; Revision 1
- AOP-37; Loss Of Instrument Air; Revision 0
- AOP-38; Acts of Nature; Revision 4

- CR-PLP-2015-05084; A-Buss and B-Buss Tripped From Their Associated Lockout Relays But Their Target Indicators Did Not Operate; Dated October 16, 2015
- CR-PLP-2015-05085; Subsequent to Closing 25R8; All 4160 Volt Busses lost Power Based On A Voltage Transient; Dated October 16, 2015
- CR-PLP-2015-05086; Need To Update Drawing E730-1; Revision 7 to Show Seal In Coil and Its Contact For Each Undervoltage Relay; Dated October 16, 2015
- CR-PLP-2015-05086; Need To Update Drawing E730-1; Revision 7 to Show Seal In Coil and Its Contact For Each Undervoltage Relay; Dated October 16, 2015
- CR-PLP-2015-05093; P-39B; "B" Cooling Tower Pump Did Not Start; Dated October 16, 2015
- CR-PLP-2015-05097; Identification and Resolution of a Grounded Conductor; Dated October 15, 2015
- CR-PLP-2015-05920; Requirements of EN-TQ-210 Conduct of Simulator Training Not Met; Dated December 1, 2015
- CR-PLP-2015-06062; Operations Managers and Supervisors Are Not Consistently Performing Post Simulator Critiques In A Manner That Fully Identifies the Most Significant Focus Areas; Dated December 8, 2015
- EN-OP-116; Infrequently Performed Tests or Evolutions; Revision 12
- EN-RE-327; PWR Startup Critical Predictions and Evaluation Process; Revision 3
- EOP Supplement 1; Pressure Temperature Limit Curves; Revision 5
- GOP-14; Shutdown Cooling Operations; Revision 50
- GOP-4; Mode 2 to Mode 1; Revision 23
- LOR Simulator Crew Performance Evaluation Report; Dated December 1, 2015
- PO-2; PCS Heatup/Cooldown Operations; Revision 7
- Simulator Exam Scenario 2015 ECPE-1; Revision 2
- SOP-1A; Primary Coolant System; Revision 28
- SOP-1C; Primary Coolant System – Heatup; Revision 21
- SOP-2A; Chemical and Volume Control System; Revision 85
- SOP-3; Safety Injection and Shutdown Cooling System; Revision 100
- SOP-6; Reactor Control System; Revision 35

1R12 Maintenance Effectiveness

- CR-HQN-2011-0879; INPO IER L2 11-2: 2009-2010 Scram Analysis; Dated August 10; 2011
- CR-PLP-1999-0279; Loss of One Out of Two Data Highways to DEH Operator Console; Dated April 16; 1999
- CR-PLP-2010-4672; Configuration Mode of Engineering Console Wouldn't Load; Dated October 27; 2010
- CR-PLP-2011-0503; Primary Power Supply for DEH OPC unit DPU 52 and OAC unit DPU 53 was Removed from Service with the Unit at Power; Dated February 2; 2011
- CR-PLP-2011-3458; MC-2000; DEH Engineer Console; Power Supply Replaced; Dated July 13; 2011
- CR-PLP-2011-3953; Review Recommendations from INPO IER L2 11-2 and Develop Corrective Actions; Dated August 11; 2011
- CR-PLP-2012-0667; Found ED-56; DEH UPS; De-energized; Dated January 31; 2012
- CR-PLP-2012-2302; Drop 52 Could Not Be Restored Following Power Supply Replacement Due to Configuration Issue with Engineering Console; Dated April 9; 2012
- CR-PLP-2012-2748; Bad Power Supply Board in ED-56; DEH UPS; Dated May 1; 2012
- CR-PLP-2012-3453; MC-2000 Hard Drive on Engineering Console Failed; Dated May 1; 2012
- CR-PLP-2014-0923; MC-2000 DEH Disk Drive Needs Replacing; Dated February 1, 2014

- CR-PLP-2014-1188; DEH UPS Battery Testing Found UPS Lasted for Less than Desired Amount of Time; Dated February 8, 2014
- CR-PLP-2014-3137; DEH Drop 254 Fault Alarm Causes Control Room Turbine Panel Trouble Alarm; Dated May 23, 2014
- CR-PLP-2014-5369; Received Control Room Turbine Panel Trouble Alarm for DEH UPS Going to Bypass; Dated November 7, 2014
- CR-PLP-2015- 4352; MC-2000, DEH Engineering Console, Power Supply Reliability; Dated September 27, 2015
- CR-PLP-2015-1358; DEH UPS on Battery and Drop 52 Fault Alarms Causes Control Room Turbine Panel Trouble Alarm; Dated April 1, 2015
- CR-PLP-2015-1364; DEH UPS on Battery and Drop 53 Fault Alarms Causes Control Room Turbine Panel Trouble Alarm; Dated April 1, 2015
- CR-PLP-2015-1622; DEH Drop 200, Engineering Station, Fault Alarm Causes Control Room Turbine Panel Trouble Alarm; Dated April 19, 2015
- CR-PLP-2015-3827; Turbine/Reactor Trip on Turbine Panel Trouble; Dated September 16, 2015
- CR-PLP-2015-3987; DEH System DPU 2 MBT Board Status LEDs Dimmer Than Others; Dated September 19, 2015
- CR-PLP-2015-4139; Drop 200 Engineering Console Power Supply is Cycling On and Off; Dated September 23, 2015
- CR-PLP-2015-4201; DEH Hardware Troubleshooting Work Order Request; Dated September 23, 2015
- CR-PLP-2015-4314; DEH Control Crossover Pressure Transmitter Found Out of As-Found Tolerances; Dated September 26, 2015
- CR-PLP-2015-4352
- CR-PLP-2015-4354; MC-2000, DEH Engineering Console, Memory Board Fault; Dated September 27, 2015
- CR-PLP-2015-4439; DEH UPS System Reliability; Dated September 29, 2015
- CR-PLP-2015-4443; DEH UPS System Incorrect Time; Dated September 29, 2015
- CR-PLP-2015-4445; DEH UPS System Incorrect UPS on Batteries Output; Dated September 29, 2015
- CR-PLP-2015-4454; DEH DPU Data highway Restoration Anomaly During Troubleshooting for WO 426051; Dated September 29, 2015
- CR-PLP-2015-4686; Found Secondary I/O Power Supply for Drop 2/52 Reading 0 Volts DC; Dated October 5, 2015
- CR-PLP-2015-4687; Printer Fault Causing Drop 200, DEH Engineers Station; Alarm; Dated October 5, 2015
- CR-PLP-2015-5152; DEH Loss of Manual Panel Did Not Cause Desired Turbine Trip Signal; Dated October 18, 2015
- CR-PLP-2015-5498; DEH System Obsolete Power Supply Refurbishment Negative Trend; Dated November 4, 2015
- CR-PLP-2015-5639; Change Out Existing DEH, OA Panel, K1-K4, K7-K-14, and K32-K34 Relays; Dated November 12, 2015
- Digital Electro-Hydraulic Controls System Health Report; Fourth Quarter 2014
- EC-60414; Panel EC-23, Eliminate DEH Turbine Trip from Overspeed Protection Control; Revision 0
- EGAD-EP-10; Palisades Maintenance Rule Scoping Document; Revision 0
- EN-DC-153; Preventive Maintenance Component Classification; Revision 12
- EN-DC-175; Single Point Failure Review Process; Revision 5
- EN-DC-203; Maintenance Rule Program; Revision 3
- EN-DC-204; Maintenance Rule Scope and Basis; Revision 3

- EN-DC-205; Maintenance Rule Monitoring; Revision 5
- EN-DC-324; Preventive Maintenance Program; Revision 15
- EN-LI-118; Cause Evaluation Process; Revision 21
- EN-OE-100; Operating Experience Program; Revision 24
- Preventive Maintenance Basis Template: EN-I&C-Electronic Circuit Cards; Revision 5
- WO 213610; EC-23, DPU and I/O (Drop 53) Power Supply Replacements; Dated October 7, 2010
- WO 383719; MC-210, DEH Operator Panel 210 in Control Room Alarm
- WO 410682; EC-23, Drop 52 Troubleshoot/Repair Highway 1
- WO 411777; MC-2000, Drop 200 Alarm on DEH Engineers Station
- WO 425161; EC-23, Drop 2 Troubleshoot and Repair
- WO 426051; EC-23, Troubleshoot DEH System for September 16, 2015 Plant Trip Root Cause
- WO 426657; ED-56, Test Old UPS for DEH System for Operation/Damage
- WO 426658; ED-56, Replace DEH UPS per EC-60574
- WO 426898; Power Supply EC-23-02A-P, Replace Drop 2 Power Supply in EC-23
- WO 430687; EC-23, Replace K1-K4, K7-K14, and K32-K34 Relays in 1R25
- WO 432194; EC-23, Replace MBD & MBT Cards for Drop 53 in 1R25
- WO 432196; MC-2000, Replace MBD & MBT Cards for Drop 200 in 1R25
- WO 432198; MC-210, Replace MBD & MBT Cards for Drop 210 in 1R26
- WO 432199; EC-23, Replace MBD & MBT Cards for Drop 3 in 1R26
- WO 432200; EC-23, Replace MBT Card for Drop 52 in 1R25
- WO 432202; EC-23, Replace MBD Card for Drop 2 in 1R26
- WO 52326110; MC-2000, DEH Engineer Console Replace Power Supply; Dated July 13, 2011
- WO 52326132; EC-23, DPU (Drop 52) Power Supply Replacement; Dated May 1, 2012
- WO 52326132; EC-23, DPU Power Supply Replacement (Drop 52); Dated May 1, 2012
- WO 52435661; EC-23, DPU (Drop 2) Power Supply Replacement; Dated February 2, 2014
- WO 52529344; DEH System Preventive Maintenance; Dated March 11, 2014
- WO 52544968; DEH Preventive Maintenance (Outage); Dated October 19, 2015
- WO 52547815; DEH System Preventive Maintenance; Dated January 6, 2015
- WT-WTPLP-2015-0272; Enhancement Actions Associated with Root Cause Evaluation for CR-PLP-2015-3827

1R13 Maintenance Risk Assessments and Emergent Work Control

- Admin 4.02; Control of Equipment; Revision 74
- Admin 4.02; Risk Management and Risk Monitoring, Attachment 3; Revision 73
- Admin 4.49; Non-Power Operation Fire Risk Management; Revision 0
- AOP-38; Acts of Nature; Revision 4
- ARP-13; 345 kV Switchyard Scheme EK-50 (C-53; C-54); Revision 55
- ARP-2; Generator Scheme EK-03 (EC-11); Revision 55
- CR-PLP-20125-06038; P/S-0550, Master Power Supply; Green Light (Normal) Indication Was Not Lit; Dated December 7, 2015
- CR-PLP-2015-00757; PT-0371, Safety Injection Tank T-82C Pressure Trans. Is Spiking Worse; Dated February 17, 2015
- CR-PLP-2015-02415; Observed PT-0371 Safety Injection Tank T-82C Pressure Trans. Jump On The PPC; Dated June 10, 2015
- CR-PLP-2015-02569; Observed PT-0371 Safety Injection Tank T-82C Pressure Trans. Change On The PPC; Dated June 22, 2015

- CR-PLP-2015-03042; Received Alarm EK-1328 Unexpectedly Due to High Pressure In T-82C; Dated July 21, 2015
- CR-PLP-2015-05681; Received alarm EK-0518; 2400V Bus 1C; 1D and/or 1E Ground Unexpectedly; Dated November 17, 2015
- CR-PLP-2015-05715; Indication of a Hotspot Was Identified At The Motor Junction Box For P-40A Dilution Water Pump; Dated November 18, 2015
- CR-PLP-2015-05720; Acrid Odor Was Noticed Coming Through the Panel Vents In Back of Switchgear Bus 1E; Breaker 152,309, Dilution Water Pump P-40B; Dated November 18, 2015
- CR-PLP-2015-05817; X-Phase Wire Has Indication of Excessive Heat; Dated November 23, 2015
- CR-PLP-2015-05967; Received Alarm EK-0518, 2400V Bus 1C, 1D and/or 1E Ground, Unexpectedly and Immediately Cleared; Dated December 2, 2015
- CR-PLP-2015-05973; Received EK-0333, Switchyard 125 VDC and 240 VAC Trouble Which Immediately Cleared; Dated December 3, 2015
- CR-PLP-2015-05974; Drawings E11-1 and E-12-1 Have Terminal Numbers To The Relay Voltage Connections for Relays 164-1, 2, and 3, That Do not Exist For These Relays; Dated December 3, 2015
- CR-PLP-2015-06013; Found Two X-Phase Cables With Splits In Their Outer Jacket; Dated December 4, 2015
- CR-PLP-2015-06064; 2400V Ground Troubleshooting Team Meeting Did Not Have An Operations Representative At The Start Of The Meeting; Dated December 8, 2015
- CR-PLP-2015-06104; Observed A Step Change In Indicated Pressure On PIA-0371; Dated December 10, 2015
- CR-PLP-2015-06130; 2400V Ground Alarm; Dated December 13, 2015
- CR-PLP-2015-06189; Observed A Step Change In Indicated Pressure On PIA-0371; Dated December 15, 2015
- Data Sheets – 2400V AC System Ground Alarms; Dated November 16, 2015 – December 13, 2015
- E-1; 400 Volt Motor Control Center Warehouse, Sheet 1; Revision 84
- E-1; Meter and Relay Diagram, Sheet A; Revision 12
- E-29; Instrument AC panel Schedule, Sheet 14B; Revision 4
- E-3; 2400 Volt system, Sheet 1; Revision 50
- EN-FAP-WM-002; Critical Evolutions; Revision 3
- EN-FAP-WM-002; Critical Evolutions; Revision 3
- EN-MA-118; Foreign Material Exclusion; Revision 10
- EN-MA-119; Material Handling Program; Revision 23
- EN-MA-125; Troubleshooting Control of Maintenance Activities; Revision 18
- EN-WM-104; On Line Risk Assessment; Revision 12
- EN-WM-105; On-Line Risk Assessment; Revision 11
- GOP-14; Shutdown Cooling Operations; Revision 49
- M-383; Miscellaneous Instrument Data Sheet, Sheet 13; Revision 6
- MSM-M-72; Movement of Heavy Loads In The Turbine Building; Revision 1
- Night & Standing Order Logs
- Operations Logs
- Palisades Shutdown Safety Risk Assessment; Dated October 9, 2015
- SOP-2A; Chemical and Volume Control system; Revision 85
- SOP-30; Station Power; Revision 77
- SOP-30; Station Power; Revision 77
- SRP-3; Electrical Auxiliaries and Diesel Generator Scheme EK-05 (EC-11); Revision 77
- TMOD 61944; P/S-0550, Install Temporary Power Supply to Allow Master Supply Replacement; Revision 0

- VEN-M201; Wiring Diagram Section C12-8, Sheet 59; Revision 65
- VEN-M201; Wiring Diagram, Section C12-4, (Sect. C12-3 Partial), Sheet 52; Revision 80
- WO 432418-01; Install New P/S-0550 Master Supply
- WO 432418-04; P/S-0550, Install Temporary Power Supply to Allow Master Power Supply Replacement

1R15 Operability Determinations and Functionality Assessments

- CR-PLP-2005-05496; Containment Sump Fill Rate Increased During QO-1; Dated October 7, 2005
- CR-PLP-2015-03859; Pencil Sized Leak Coming from the Storz Connection Cap; Dated September 16, 2015
- CR-PLP-2015-04251; Piping Could Not Be Drained Due to Leak by of CV-3057; Dated September 25, 2015
- CR-PLP-2015-04486; Water and Corrosion Found in Containment Building Floor Liner Leak-Chase Channels; Dated September 29, 2015
- CR-PLP-2015-04494; Conditions Observed While Performing RO-216; Dated September 29, 2015
- CR-PLP-2015-04504; During RO-216, Service Water Flow To EDG 1-1 Did Not Meet The Acceptance Criteria; Dated September 30, 2015
- CR-PLP-2015-04616; Inservice Test of the Shutdown Cooling Control Valves Testing Frequency will Expire; Dated October 3, 2015
- CR-PLP-2015-04626; Inservice Test of Component Cooling Water Pump P-54C Could not be Performed; Dated October 3, 2015
- CR-PLP-2015-04933; During Review of FSAR Table 9-1, Discovered a Misnomer for Service Water System Flow Requirements Listed for The Instrument Air Compressors; Dated October 11, 2015
- CR-PLP-2015-04973; Service Water flow to 1-1 EDG Was low OOS IAW; Dated 10/13/2015
- CR-PLP-2015-05095; Pinhole Leak In The Service Water System Inside of Containment; Dated October 16, 2015
- Design Basis Document 2.03; Containment Spray System; Revision 9
- EA-C-PAL-99-1209B-01; Generation of Flow Rate Acceptance Criteria for Technical Specification Surveillance Test RO-216; Revision 3
- EC 60770; Change Class 3 Boundary for Fire Water In Containment Building; Revision 9
- EC Mark-Up for #60770; M-208, Component Cooling Water Heat Exchangers, Sheet A; Revision 26
- EC Mark-Up for #60770; M-208, Control Room HVAC Condensers, Sheet 1B; Revision 39
- EN-DC-128; NFPA 805 Preliminary Risk Screening for EC60770; Revision 10
- EN-DC-128; NFPA 805 Qualitative Risk Evaluation for EC60770; Revision 10
- EN-LI-100; EC 60770, Change Class 3 Boundary for Fire Water In Containment Building; Revision 16
- EN-OP-104; Operability Determination Process for CR-PLP-2015-05095; Revision 9
- FPIP-4; Fire Protection Systems and Fire Protection Equipment; Revision 35
- FPSP-MO-1; Fire Suppression Water System Valve Alignment Verification Checkoff Sheet; Revision 21
- FPSP-RO-8; Containment Building Fire Hose Replacement, Nozzle Inspection, and Station Valve Check; Revision 5
- FSAR Chapter 5; All Other Components, Table 5.2-3; Revision 31
- FSAR Chapter 5; Interconnecting Piping and Valves Required to Service Quality Group C System Components, Table 5.2-3; Revision 31

- M-208; Service Water System, Sheet 1A; Revision 64
- M-208; Service Water System, Sheet 1B; Revision 37
- M-208; Service Water System, Sheet A; Revision 26
- M-208; Sheet 1B; Revision 37
- PLP-RPT-15-00030; Evaluation of 1R24 Flow Balancing Results for Cycle 25; Revision 0
- QO-42; Inservice Testing of Shutdown Cooling Control Valves; Revision 18
- RO-216; Service Water flow Verification; Revision 21
- RT-71K; Technical Specification Surveillance - Class 2 System Functional Test for Shutdown Cooling System; Revision 10
- SEP-APJ-PLP-01; PLP Mechanical Containment Penetrations Basis Program; Revision 1
- SEP-ISI-PLP-002; ASME Code Boundaries for ASME Section XI, P&ID M-208, Sheet 1B, Service Water; Revision 1
- SEP-ISI-PLP-002; PLP ASME Code Boundaries for ASME Section XI Inservice Inspection Program; Revision 2
- SOP-15; Service Water System Checklist – Critical; Revision 60

<u>1R18 Plant Modifications</u>

- CR-PLP-2015-04900; Core Exit Thermocouple 30 is Reading Low; Dated October 10, 2015
- CR-PLP-2015-05003; One Phase of a Newly Installed 1000 MCM Cable Failed Tan Delta Testing; Dated October 14, 2015
- CR-PLP-2015-05016; Create Work Order to Install TMOD EC-60676; Dated October 14, 2015
- CR-PLP-2015-05018; Four Core Exit Thermocouples Appear to be Invalid; Dated October 14, 2015
- Design Basis Document 3.04; 2400V AC System; Revision 8
- EN-DC-115; Engineering Change Process; Revision 17
- MT-10; Core Monitoring; Revision 18
- Permanent Modification EC-55367; Replace 1C and 1D 500MCM Feeder Cables from Startup Transformer 1-2 with 1000MCM Feeder Cables
- PO-3; Alternate Incore and Excore Applications; Revision 7
- RFL-D-12; Reactor Head Electrical Cable Disconnection; Revision 11
- RFL-D-15; Instrument Nozzle Flange Removal; Revision 8
- SOP-30; Station Power; Revision 8
- Temporary Modification EC-60676; Swap Qualified Incore Cables Locations with Non-Qualified Cables to Ensure 4 Qualified Circuits per Reactor Quadrant Following R24; Dated October 15, 2015
- WO 00396134-08; EX-04 Tan Delta Testing of Installed Cabling; Dated October 14, 2015
- WO 396134-08; 1C and 1D Tan Delta Cable Test; Dated October 9, 2015
- WO 51619208-02; PM on EX-04 Cables to Busses C, D, and E; Dated October 16, 2015

<u>1R19 Post-Maintenance Testing</u>

- CRD-E-1; Removal, Zeroing, And Installation of CRDM Drive Packages on the Reactor Vessel; Revision 25
- CRD-M-13; CRDM Seal Housing Replacement Procedure; Revision 32
- CRD-M-31; Rebuilding and Testing CRDM Seal Housing Assemblies; Revision 19
- CR-PLP-2011-00693; MV-CVC2212 P-55C, MV-CVC2307 P-55C, MV-CVC2206 P-55C Are All Leaking By; Dated February 11, 2011
- CR-PLP-2013-00556; MV-CVC2212, P-55C Manifold Flush Failed PMT For Work Order 266191; Dated February 8, 2013

- CR-PLP-2014-02004; P&ID M202 Sheet 1B Does Not Provide A Line Specification for the 3 Charging Pump Discharge Manifold Flush Inlet Pipes; Dated March 9, 2014
- CR-PLP-2014-03449; CRD-27 SPI Erroneous Indication; Dated June 21, 2015
- CR-PLP-2014-05824; A Small leak Was Found Downstream of PCV-2288, N_2 Station 7 Pressure Control Valve
- CR-PLP-2015-00920; QO-5, Valve Test Procedure Currently Does Not Give Specific Guidance for Operations To Assess The Technical Specification Operability of Safety Related Valves; Dated February 27, 2015
- CR-PLP-2015-01181; During MO-7A-2, Emergency Diesel Generator 1-2 Failed to Start; Dated March 15, 2015
- CR-PLP-2015-04022; CV-3057 Did Not Fully Open As Indicated Locally At the Valve; Dated September 2, 2015
- CR-PLP-2015-04022; One of CV-3057 Solenoid Valves Is Not Functioning Properly, Preventing CV-3057 Proper Operation; Dated September 20, 2015
- CR-PLP-2015-04033; Air Leak Heard Around Cv-3057; Dated September 20, 2015
- CR-PLP-2015-04043; Air Leak Heard Around CV-3057, SIRW Tank Outlet Isolation Valve; Dated September 20, 2015
- CR-PLP-2015-04462; Adjustment Needed on PCV-3057B, T-58 Outlet CV-3057 A/S REG Setpoint; Dated September 29, 2015
- CR-PLP-2015-04462; RV-3057B Opening Pressure To CV-3057 Was Blowing By; Dated September 29, 2015
- CR-PLP-2015-04501; Regulator PCV-3057B Is Leaking Externally Through The Diaphragm and Needs to be Replaced; Dated September 30, 2015
- CR-PLP-2015-04501; Replace PCV-3057B; Dated September 30, 2015
- CR-PLP-2015-04787; CRDM Support tube #31 Was Not Properly Seated In The Upper Housing; Dated October 7, 2015
- CR-PLP-2015-05031; Evidence Of leakage From the Packing Gland of CV-3057; Dated October 15, 2015
- CR-PLP-2015-05031; Evidence of Leakage From The Packing Gland Of CV-3057, SIRW Tank T-58 Outlet Isolation; Dated October 15, 2015
- CR-PLP-2015-05058; Air Leak Remains on CV-3057 After Repairs; Dated October 15, 2015
- CR-PLP-2015-05058; Documenting Current Condition/Status of CV-3057; Dated October 15, 2015
- CR-PLP-2015-05839; NRC Resident Questioned the RV-2279 “N2 Station 5 PCV-2270 Outlet Relief” Setpoint Value; Dated November 24, 2015
- CR-PLP-2015-05894; During QO-5 Valve Stroke of CV-3018 Acrid Odor Noticed; Dated December 1, 2015
- CR-PLP-2015-05895; During Performance of QO-5 Valve Strokes, SV-3059 was Blowing By; Dated December 1, 2015
- CR-PLP-2015-05933; The Mid-Cycle Team Made The Observation That QO-5X Had An Integrated Risk Level of Medium; Dated December 1, 2015
- CR-PLP-2015-06005; P-54A D/P Indicator Isolation Handle Fell Off
- CR-PLP-2015-06028; The Work Instructions of Work Order 52570509-01 Insufficiently Instructed Repairmen to Perform Steps 5.4.1 Through 5.4.15 of EN-MA-134; Dated December 4, 2015
- CR-PLP-2015-06214; MV-CVC2212 Was Tested to a Different Pressure Than After Previous Replacements; Dated December 17, 2015
- CR-PLP-2015-4227; After Installing the New Seal Package on P-50C, Discovered an O-Ring was Not Installed As Required; Dated September 24, 2015
- CR-PLP-2015-4593; Measured Tolerance of New P-50C Shaft Seal Cartridge Exceeds Acceptance Criteria per Maintenance Procedure; Dated October 2, 2015

- CVCO-4; Periodic Test Procedure – Charging Pumps; Revision 8
- EC 31817; Revise the Hydraulic Pipe-Flo Model for the ESS from Version 4.11 to Version 2007A, Revise the ESS Pump Curve Calculation, and Recirculation Mode NPSH Calculation; Revision 0
- EC 49790; Update P&ID M-202 Sheet 1B to Provide Piping Line Class Information for Charging Pump P-55A/B/C Discharge Manifold flush Inlet Lines; Revision 0
- EC 60353; Replace PCV-3057B, T-58 Outlet CV-3057 Air System Regulator, with A Different Style Regulator; Revision 0
- EC-60353; Replace PCV-3057B, T-58 Outlet CV-3057, Air System Regulator With a Different Style Regulator; Revision 0
- EN-MA-101; Conduct of Maintenance; Revision 18
- EN-MA-118; Foreign Material Exclusion; Revision 10
- EN-MA-134; Offline Motor Electrical Testing; Revision 5
- EN-WM-105; Planning; Revision 16
- M-202; Piping & Instrument Diagram, Chemical & Volume Control System, Sheet 1B; Revision 59
- M-203; Safety Injection Containment Spray and Shutdown Cooling System, Sheet 2; Revision 28
- M-203,Safety Injection; Containment Spray, and Shutdown Cooling System, Sheet 2; Revision 28
- M-204; Safety Injection Containment Spray & Shutdown Cooling System, Sheet A; Revision 8
- M-204; Safety Injection Containment Spray and Shutdown Cooling System, Sheet 1; Revision 86
- M-204; Safety Injection Containment Spray and Shutdown Cooling System, Sheet 1A; Revision 44
- M-204; Safety Injection Containment Spray and Shutdown Cooling System, Sheet 1B; Revision 41
- M-209; Component Cooling System, Sheet 2; Revision 33
- M-212; Instrument Air Walkdown, Sheet 4; Revision 30
- M-222; Miscellaneous Gas Supply Systems, Sheet 2; Revision 30
- M-225; High Pressure Air Operated Valves, Sheet 16; Revision 6
- M-259; Piping class Summary; Revision 26
- M-347 Sheet 39; "Pressure Safety (Relief) Valve Data Sheet," Revision 4
- MO-7A-2; Emergency Diesel Generator 1-2; Revision 87
- MSM-M-40; Safety/Relief Valve Testing; Revision 15
- PCS-M-52B; Testing the N-9000 Series PCP Seal Cartridge with the Stainless Steel Test Fixture with Backpressure Capability; Revision 0
- PCS-M-53; Removal and Installation of N-9000 Series Primary Coolant Pump Shaft Seal Cartridge; Revision 23
- PCS-M-54; N-9000 Primary Coolant Pump Shaft Seal Assembly; Revision 12
- PCS-M-55; Disassembling and Inspecting N-9000 Primary Coolant Pump Shaft Seal Cartridge; Revision 3
- QO-16 Basis Document; Inservice Test Procedure – Containment Spray Pumps; Revision 16
- QO-16; Inservice Test Procedure – Containment Spray Pumps; Revision 35
- QO-5; Basis Document for QO-5, Valve Test Procedure (Includes Containment Isolation Valves); Revision 18
- RO-22; Control Rod Drop Times; Revision 21
- RO-22; Control Rod Drop Times; Revision 21
- SEP-PLP-IST-101; "Inservice Testing of Plant Valves;" Revision 1
- SOP-19; Instrument Air System Nitrogen/Air Backup Stations; Revision 64
- WO #396557; P-50C Seal Rebuild and Testing

- WO 265920; MV-CVC2206, P-55C Discharge Manifold Flush Outlet Leaks By Seat
- WO 266192; MV-CVC-2307, P-55C Discharge Manifold flush Outlet Leaks By Seat
- WO 280353-03; CRD-02, CRD Seal Housing, Rebuild/Test During Outage
- WO 344577; Valve MV-CVC2212 Failed Its PMT and Needs Rebuilt/Replaced
- WO 344578; Valve MV-CVC2206 Failed Its PMT and Needs Rebuilt/Replaced
- WO 380851-05; “NSD, Return To Service Testing”
- WO 387455; High Pressure Air Low pressure Alarm When Opening CV-3057
- WO 387455; High Pressure Air Low Pressure Alarm When Opening CV-3057
- WO 387759-01; RSPT-14
- WO 395880-01; CRD-26, “CRDM Seal Housing”
- WO 396082-01; CRD-37, “CRDM Seal Housing”
- WO 426538; Replace PCV-3057B
- WO 426538; Replace PCV-3057B
- WO 427803; CV-3057, Air Leak At Stem On Clevis End
- WO 52434882-01; EEQ Maint: VOP-3007, Condition Check
- WO 52434882-02; EEQ Maint: VOP-3007, Condition Check
- WO 5250687 04; RV-2279, OPS PMT
- WO 52506877 02; RV-2279, Test Replacement Relief
- WO 52506877; RV-2279 Mechanical
- WO 52538662 01; RO-22, Control Rod Drop times
- WO 52538662; RO-22 – Control Rod Drop Times
- WO 52556705-01; EMA-1210, P-54A Motor Bearing Oil Change (Electrical) *EEQ*
- WO 52570509-01; EEQ – EMA-1210 Containment Spray P-54A Motor
- WO 52647151-01; QO-16A – P-54A, IST Containment Spray Pump
- WO 52647152-01; QO-5 Valve Test Procedure (Includes Containment Isolation Valve)
- WO-427803; CV-3057, Air Leak At Stem On Clevis End

<u>1R20 Outage Activities</u>

- AOP-25; “Loss of Refueling Water Accident;” Revision 0
- AOP-26; “Loss of Spent Fuel Pool Cooling;” Revision 2
- AOP-30; “Loss of Shutdown Cooling;” Revision 1
- AOP-34; “Fuel Handling Accident;” Revision 0
- C-0246; Reactor Building Safety Injection Tank supports, Detail 5; Revision 4
- CF-PLP-2015-03892; MO-3012, LPSI To Reactor Coolant Loop 2A Was Discovered to Have Dry Packing and Body To Bonnet Boric Acid Leaks; Dated September 16, 2015
- CR-2015-05026; Evidence of Leakage from the Packing Gland of MV-ES103; Dated October 15, 2015
- CRD-M-13; CRDM Seal housing Replacement Procedure; Revision 32
- CR-PLP-2007-04879; Evaluate Co-Ordination of Main Transformer (EX-10) Cooling Breakers; Dated June 18, 2007
- CR-PLP-2012-03407; EX-10 (Main Transformer) Cooling Lost While In Backfeed Mode; Dated April 29, 2012
- CR-PLP-2012-03971; Breakers Need To Be Ordered and Work Order Scheduled Without Impacting T-28 Schedule; Dated October 3, 2012
- CR-PLP-2012-03971; EK-0325, Breakers 8-2 and 8-10 Have Tripped; Dated May 18, 2012
- CR-PLP-2014-02116; Primary Coolant Pump P-50B Noted To Be Spritzing Drops of Water; Dated March 14, 2014
- CR-PLP-2015,04576; MV-CVC2225 Letdown Orifice Bypass Stop Valve Has A Significant Quantity Of Dark Red Boric Acid; Dated October 2, 2015

- CR-PLP-2015-03830; CV-0909 Letdown HX E-58 CCW Outlet Has A Packing Leak; Dated September 16, 2015
- CR-PLP-2015-03856; White Crystalline Substance Discovered at the Base of the Tool Access Flange On CRD-36 Seal Housing; Dated September 16, 2015
- CR-PLP-2015-03888; MV-Es3011, HPSI TRN 1 Loop 2A MO-3011 Inlet Was Discovered to Have A Dry; White Minor Packing Boric Acid Leak; Dated September 17, 2015
- CR-PLP-2015-03891; MO-3007, HPSI To Reactor Coolant Loop 1A Train Was Discovered To Have A Dry Boric Acid Leak From The Valve Packing; Dated September 17, 2015
- CR-PLP-2015-03893; MO-3008 LPSI To Reactor Coolant Loop 1A Was Discovered to Have Dry Packing and Body To Bonnet Boric Acid Leaks; Dated September 17, 2015
- CR-PLP-2015-03894; MO-3010 LPSI To Reactor Coolant Loop 1B Was Discovered To Have Dry Packing and Body To Bonnet Boric Acid leaks; Dated September 17, 2015
- CR-PLP-2015-03898; MO-3049, Safety Injection Tank T-82C outlet Isolation Valve Was Discovered To Have A Dry Boric Acid Leak From The Valve Packing; Dated September 17, 2015
- CR-PLP-2015-03900; MO-3062 HPSI Train 2 Loop 2B Was Discovered To Have A Dry Boric Acid Leak From The Valve Packing; Dated September 17, 2015
- CR-PLP-2015-03910; Mode 3 Walkdown, MV-SFP505, Reactor Cavity Drain To Sump Was Discovered To Be Covered With Dry Boric Acid; Dated September 17, 2015
- CR-PLP-2015-03912; Mode 3 Walkdown Of CRD-5 Has Dry Boric Acid leak; Dated September 17, 2015
- CR-PLP-2015-03913; CRD-18 Control Rod Drive Mechanism Was Discovered To Have A Dry Boric Acid Leak; Dated September 17, 2015
- CR-PLP-2015-03918; During Mode 3 Walkdown of the PCS, the Concrete Near The Bottom of the Pressurizer Was Discovered To Have Dry Boric Acid On It; Dated September 18, 2015
- CR-PLP-2015-03957; Mode 3 Walkdown – Bio Shield Penetration; Dated September 16, 2015
- CR-PLP-2015-03959; FS-2104, Reactor Shield Cooling Coils Flow Switch Discovered To Have CCW Leakage On A Flange and Pipe Below; Dated September 16, 2015
- CR-PLP-2015-03982; Potential Work Hours Violations Not Flagged In PQ and S; Dated September 19, 2015
- CR-PLP-2015-04036; Waiver For the Fatigue Management Program; Dated September 20, 2015
- CR-PLP-2015-04060; Boron Indications Were Identified Below the CRDM Drive Motor Flange On CRDM #5 and #36; Dated September 21, 2015
- CR-PLP-2015-04092; CV-2099 Failed Drop Test; Dated September 22, 2015
- CR-PLP-2015-04119; Motor Lead Wires for MO-1043A Found to Be Degraded; Dated September 22, 2015
- CR-PLP-2015-04120; A Mechanical Maintenance Supervisor Inadvertently Worked 72.5 Hours In A Rolling 7 Day Period; Dated September 22, 2015
- CR-PLP-2015-04130; “B” Channel AFAS Poser Supply Appears To Be Back; Dated September 22, 2015
- CR-PLP-2015-04139; ’B’ Channel AFAS Power Supply Appears To Be Bad; Dated September 22, 2015
- CR-PLP-2015-04142; Failure to Inspect and/or Increase Torque On North End of E-19, Turbine Gland Seal Condenser; Dated October 18, 2015
- CR-PLP-2015-04167; A Work Hour Waiver is Required for Covered Workers in the Mechanical Maintenance Department; Dated September 23, 2015
- CR-PLP-2015-04203; CV-0780 Failed A Drop Test Per EN-MA-143; Dated September 24, 2015
- CR-PLP-2015-04204; CV-0780 Failed A Drop Test Per EN-MA-143; Dated September 24, 2015

- CR-PLP-2015-04216; Work Hour Waiver Is Required For Covered Workers In the Mechanical Maintenance Department; Dated September 24, 2015
- CR-PLP-2015-04246; P-50A Has An Active Leak From the Gasket for The Lower Bearing Reservoir; Dated September 25, 2015
- CR-PLP-2015-04298; Documentation of the As Left Status of MO-1043A; Dated September 25, 2016
- CR-PLP-2015-04357; Steam Leak Path Identified During Inspection of the K-1 HP, High Pressure Turbine; Dated September 27, 2015
- CR-PLP-2015-04362; Issues Found While Working Under WO 380851-02 For Cleaning Of the CRD Seal Housings; Dated September 27, 2015
- CR-PLP-2015-04372; Investigate Condition Regarding Motor Lead Wires for MO-1043A; Dated September 27, 2015
- CR-PLP-2015-04411; A Dry Residue Of Boric Acid Was Observed On Piping Underneath CV-1103; Dated September 28, 2015
- CR-PLP-2015-04488; MV-Es3184, HPSI P-66A Suction Manual Valve Was Difficult to Operate; Dated September 29, 2015
- CR-PLP-2015-04489; Flex 400 kw Generator Would Not Start; Dated September 29, 2015
- CR-PLP-2015-04490; Fuel Movement Deviation Form Performed on The Core Shuffle Move Sheets Did Not Include the Correct Orientation for Control Rod 134; Dated September 29, 2015
- CR-PLP-2015-04506; Computer Errored and Would Not Allow Lowering of the Grappled Control Blade During Inspections of them; Dated September 30, 2015
- CR-PLP-2015-04519; The Loading of New Fuel Canisters Onto A Short Flatbed Truck Was Not Performed While Place-keeping Step5.2.5 In FHS-M-10 “New Fuel Receipt;” Dated September 30, 2015
- CR-PLP-2015-04533; CV-3038Threads On Valve Actuator Stem Were Sheared; Dated September 30, 2015
- CR-PLP-2015-04538; Design Drawing vs As-Built configuration Discrepancies Found on Safety Injection Bottle/Tank T-82C; Dated September 30, 2015
- CR-PLP-2015-04539; Issues With communication Equipment and Computer Programs Were Experienced During fuel Moves; Dated September 30, 2015
- CR-PLP-2015-04559; Breaker 252-201 Had Signs Of Extensive Heat Damage On All Phases; Dated October 2, 2015
- CR-PLP-2015-04571; Unable to Perform Diagnostic Testing on CV-3056, SIRW Tank T-58 Recirc; Dated October 1, 2015
- CR-PLP-2015-04580; Improper PPE Was Briefed and Used While Racking A Breaker On a De-energized Bus; Dated October 2, 2015
- CR-PLP-2015-04581; Loss of EX-10 (Main Transformer) Cooling During Post Modification Testing; Dated October 2, 2015
- CR-PLP-2015-04583; Gasket Between Lower Bearing Cover and Oil Reservoir Short, Leaving A Gap; Dated October 2, 2015
- CR-PLP-2015-04584; Object Noted Floating in The Reactor Cavity; Dated October 2, 2015
- CR-PLP-2015-04595; CV-1057 Failed Pre-Installation Air Leak Test; Dated October 2, 2015
- CR-PLP-2015-04596; Several Pieces of Flexitallic Gasket Found On the Periphery of Tube Bundle During FOSAR In Steam Generator A; Dated October 2, 2015
- CR-PLP-2015-04601; CV-0555 and CV-0539 Were Steam Cut and Leaked By; Dated October 2, 2015
- CR-PLP-2015-04602; CV-0611, Heater E-4A Drain Has Worn Lower Stem and Requires Replacement; Dated October 2, 2015
- CR-PLP-2015-04611; Breaker Tried To Charge Charging Springs When Closed; Dated October 3, 2015

- CR-PLP-2015-04614; CV-2099, Controlled Bleedoff Isolation, Green Closed Light Would Not Actuate in the Control Room; Dated October 3, 2015
- CR-PLP-2015-04617; A Worker In FMEZ1 Had Items Not Necessary for the Job Attached To his Identification Badge Lanyard; Dated October 3, 2015
- CR-PLP-2015-04627; Work on Containment Air Cooler VHX-4 Service Water Outlet Check Valve CK-SW410 Deferred to Outage 1R25; Dated October 3, 2015
- CR-PLP-2015-04628; CV-3046 Failed Its PMT, Open Position Indication Did Not Work; Dated October 3, 2015
- CR-PLP-2015-04641; Air Line To CV-2099, PCP Controlled Bleed Off Containment ISOL is Damaged; Dated October 3, 2015
- CR-PLP-2015-04644; CV-0710, Feed pump P-1B Recirculation, Unable to Be Disassembled; Dated September 29, 2015
- CR-PLP-2015-04648; CV-0703, E-50B Feed Regulating Valve Is Missing Stem Nut; Dated October 4, 2015
- CR-PLP-2015-04650; 3 MO-1043A Motor Leads Were Found To Be Much Worse Than Expected; Dated October 4, 2015
- CR-PLP-2015-04659; 3 Screws Discovered During the Cleaning of the Containment Sump and Removed; Dated October 4, 2015
- CR-PLP-2015-04660; R-clip Discovered Loose At the Base of F-1055 During Containment Sump Envelope Inspection; Dated October 4, 2015
- CR-PLP-2015-04667; CV-0909 Has Stem Wear That Can Be Contributing To The Previous Packing Leaks; Dated October 4, 2015
- CR-PLP-2015-04676; Handwheel Broke At The Stem Closing MV-CC105; Dated October 4, 2015
- CR-PLP-2015-04679; CV-0616 ‘Heater E-2A Drain’ Failed Its Drop Test; Dated October 4, 2015
- CR-PLP-2015-04684; Transmitter found Out Of Tolerance During RI-24 B, Steam Generator Feedwater Flow Instrument loop Calibration; Dated October 5, 2015
- CR-PLP-2015-04689; Mobile Crane Checklist from EN-MA-119, Material Handling Program Was Not Completed By Electrical Maintenance Vendor on Startup Transformer Work; Dated October 5, 2015
- CR-PLP-2015-04692; CRD-M-31 Seal Housing Needs Rebuilding; Dated October 5, 2015
- CR-PLP-2015-04693; HGR/DBB1-H50 Found To Be Lightly Loaded and Loose; Dated October 5, 2015
- CR-PLP-2015-04698; Total Weight of Sodium Tetraborate Basket Weights Is Less Than The Required Minimum Weight; Dated October 5, 2015
- CR-PLP-2015-04710; MV-CC105, Letdown Heat Exchanger, E-58 CCW Inlet, Has A Broken Operator Handle; Dated October 5, 2015
- CR-PLP-2015-04713; A Negative Trend In Actuator/Valves Performance Noted For CV-0869; Dated October 5, 2015
- CR-PLP-2015-04716; Boric Acid Residue Found on Differential Pressure Transmitter Bolts, Nuts, and Mounting Brackets; Dated October 5, 2015
- CR-PLP-2015-04725; The Integrated Leak Rate Testing Team Established Work Area Does Not Maintain System Cleanliness Requirements for FME Control; Dated October 5, 2015
- CR-PLP-2015-04729; Dose of 863 mrem Accrued Investigating and Lifting Individual CRD Support Tubes to Free Pinched FME Covers; Dated October 6, 2015
- CR-PLP-2015-04731; Small Piece of Plastic Locking Tab on Draw String for FME Cover on Seal Housing for CRDM 18 Is Missing; Dated October 6, 2015
- CR-PLP-2015-04735; Support Tube For CRDM 28 Is Not Inserted Into The Keyway Of The CRDM Housing; Dated October 6, 2015

- CR-PLP-2015-04737; Stem Bushing From Stock Too Small For MV-CVC2101; Dated October 6, 2015
- CR-PLP-2015-04740; CK-ES3166 Is Extremely Hard To Operate; Dated October 6, 2015
- CR-PLP-2015-04742; Valve and Valve Actuator Ordered for CV-0608 Is Not Correct; Dated October 6, 2015
- CR-PLP-2015-04750; CRD-14 Has Degraded Spider Coupling; Dated October 6, 2015
- CR-PLP-2015-04761; Containment Box Was Brought Into The RCA With Lose Sand and Gravel On The Bottom; Dated October 7, 2015
- CR-PLP-2015-04767; Foreign Material Discovered Inside The Bonnet Area of MV-CVC2225; Dated October 7, 2015
- CR-PLP-2015-04771; Administrative Issue with Performance of IPTE Briefs, Failure to Perform Some Sections As Required; Dated October 7, 2015
- CR-PLP-2015-04787; CRDM Support Tube #31 Was Not Properly Seated in the Upper Housing; Dated October 7, 2015
- CR-PLP-2015-04791; Work Stopped Due to Incorrect Reassembling of Valve During Work on Main Turbine Stop Valve #2; Dated October 7, 2015
- CR-PLP-2015-04792; Work Stopped Due to Incorrect Reassembling of Valve During Work on Main Turbine Stop Valve #2; Dated October 7, 2015
- CR-PLP-2015-04801; Watchbill Violation For A Radiation Protection Technician; Dated October; 8, 2015
- CR-PLP-2015-04805; Half A Flannel Shirt Found During Removal of Cooler From Startup Transformer EX03; Dated October 8, 2015
- CR-PLP-2015-04838; Valve Disc and Seat Eroded and Pitted on MO-3066, HPSI Pump P-66A; Dated October 8, 2015
- CR-PLP-2015-04853; 18 Inch Torque Wrench and 5/8 Socket Dropped on RX Head; Dated October 9, 2015
- CR-PLP-2015-04855; Pressurizer Manway Average Torque Value Exceeds The Upper Range Limit; Dated October 9, 2015
- CR-PLP-2015-04858; An Employee Was Observed Attempting To Climb Over A Missile Shield Support During The Reactor Head Lift Without Adequate Fall Protection; Dated October 9, 2015
- CR-PLP-2015-04859; ICI 2-5 Is Approximately Six Inches Too High to Be Fully Inserted During Installation at Flange Location; Dated October 9, 2015
- CR-PLP-2015-04863; Work Hour Waiver Is Required For Covered Workers In the Projects Department; Dated October 9, 2015
- CR-PLP-2015-04865; Items To Be Addressed For Long Term Health and Integrity of Containment Sump; Dated October 9 2015
- CR-PLP-2015-04867; Covered Worker Exceeds Work Hour Controls While Performing Non-Covered Work; Dated October 9, 2015
- CR-PLP-2015-04878; Three Mechanical Repairpersons Were Not Signed On To Appropriate Tagging During CRDM Seal Housing Installation; Dated October 9, 2015
- CR-PLP-2015-04879; Five Safety Related CRDM Drive Motor Mounting Flanges Were Lost; Dated October 9, 2015
- CR-PLP-2015-04889; Seating Washer on SV-0507B Is Bent And Allowing Air to Leak By; Dated October 10, 2015
- CR-PLP-2015-04901; Initial Rigging for P-2A Condensate Pump Motor Was Too Long For The Load To Clear The Moister Separator/Reheater Piping; Dated October 10, 2015
- CR-PLP-2015-04902; Maintenance Support Personnel Reached Inside the P-2A Condensate Pump Motor While It Was A Suspended load Through An Uncovered Motor Vent; Dated October 10, 2015

- CR-PLP-2015-04903; Oil Leaking From 4 of 6 RTD Wells On Primary Coolant Pump Motor P-50D; Dated October 10, 2015
- CR-PLP-2015-04909; Create Work Order To Internally Inspect The Entire Length of The Intake Pipe and To Inspect The Lakebed Immediately Above The Intake Pipe; Dated October 10, 2015
- CR-PLP-2015-04916; Z-Phase Relay On Breaker 152-210, Containment Spray Pump P-54A, Had Its Seal In Pick Up Above Setpoint of 2.0A; Dated October 11, 2015
- CR-PLP-2015-04922; CRD Coupling Difficulties During 1R24; Dated October 11, 2015
- CR-PLP-2015-04940; Install/Remove A Temporary Valve Stop Block on CV-0605; Dated October 12, 2015
- CR-PLP-2015-04941; Install/Remove A Temporary Valve Stop Block On CV-0601; Dated October 12, 2015
- CR-PLP-2015-04945; Door-141 Was Found With Only One Dog Dogged; Dated October 12, 2015
- CR-PLP-2015-04955; Tan Delta Testing and Withstand Testing of 2.4kV Bus 1E Feeder Cables from Startup Transformer 1-2 Indicated Action Required; Dated October 12, 2015
- CR-PLP-2015-04964; Stroke Close Time for CV-2099 Is Close To Upper Limit of Acceptable Stroke Time; Dated October 13, 2015
- CR-PLP-2015-04984; Oil Leaking From Upper Spring Assembly on Snubber S2MSS-SGA-REF1-SS-7; Dated October 13, 2015
- CR-PLP-2015-04986; LS-0160 Reactor Vessel Flange Leak Drain Level SW Is Tripped; Dated October 13, 2015
- CR-PLP-2015-04991; K-2, Main Turbine Turning Gear Won’t Disengage; Dated October 14, 2015
- CR-PLP-2015-04997; QO-5 Closed Stroke Time of CV-0770 Was 13, 1 Seconds, Outside of the Acceptable Stroke Time Range; Dated October 14, 2015
- CR-PLP-2015-05002; CV-2004, Letdown Orifice Stop Does Not Indication Open; Dated October 14, 2015
- CR-PLP-2015-05006; N2, Station #7 Has Excessive Leakage; Dated October 14, 2015
- CR-PLP-2015-05007; NRC Identified Housekeeping Concerns In the Auxiliary Building; Dated October 14, 2015
- CR-PLP-2015-05013; Reinstall the Debris Screen On containment floor Drain F-1063; Dated October 14, 2015
- CR-PLP-2015-05014; 100% Inspection of 8 Cold Leg Welds Was Not Attainable; Dated October 14, 2015
- CR-PLP-2015-05024; Evidence of Leakage From the Packing Gland of MV-ES3417; Dated October 15, 2015
- CR-PLP-2015-05025; Evidence of Leakage From the Packing Gland of MV-ES3210; Dated October 15, 2015
- CR-PLP-2015-05027; Evidence of Leakage From the Packing Gland of MV-ES3178; Dated October 15, 2015
- CR-PLP-2015-05028; Evidence of Leakage From the Packing Bland of MV-ES3413; Dated October 15, 2015
- CR-PLP-2015-05029; Evidence of Leakage From the Pipe Cap of MV-ES3415; Dated October 15, 2015
- CR-PLP-2015-05030; Evidence of Leakage From the Packing Gland of MV-ES3231; Dated October 15, 2015
- CR-PLP-2015-05032; Evidence Of Leakage From the Packing Gland of CV-3029; Dated October 15, 2015
- CR-PLP-2015-05056; Dampness Reported on CRD# 29; Dated October 15, 2015

- CR-PLP-2015-05066; PDIL and PPDIL Check And Control Rod Out-Of-Sequence Alarm; Dated October 16, 2015
- CR-PLP-2015-05070; Group 2 Backup heaters For The Pressurizer Do Not Display A Red Light When Turned on; Dated October 16, 2015
- CR-PLP-2015-05071; WO 396134; “EX-04 Reinstall Protector Wire On Deluge Pipe” Did Not Include Enough Detail to Meet Requirements; Dated October 16, 2015
- CR-PLP-2015-05078; Bettis Operator Has Shifted and Is Misaligned, And Is Unable To Be Placed In Auto Operation; Dated October 16, 2015
- CR-PLP-2015-05080; During Performance of MI-43 Reactor Vessel Level Monitoring system Channel Check, Two Sensors in the “B” Channel Were Not Responding Properly; Dated October 16, 2015
- CR-PLP-2015-05090; Nitrogen Stations #3B and #2 Are Outside The Preferred Band; Dated October 16, 2015
- CR-PLP-2015-05091; Fatigue Waiver did Not Contain An Expiration Date; Dated October 16, 2015
- CR-PLP-2015-05094; Ventilation Started During Replacement of CRD Seal Package #29, Causing Many Items To Blow Around the Head Are; Dated October 16, 2015
- CR-PLP-2015-05096; Wide Range Steam Generator Levels, LI-0757A&B, Do Not Agree and Require Backfill; Dated October 16, 2015
- CR-PLP-2015-05098; CV-0152 Has Nitrogen Leak Through Packing; Dated October 16, 2015
- CR-PLP-2015-05100; P-54C Containment Spray Pump Received EK-1161 Alarm; Dated October 17, 2015
- CR-PLP-2015-05101; Unexpected alarm EK-1347 Containment Air Coolers Service Water Leak; Dated October 17, 2015
- CR-PLP-2015-05102; Evident Leakage from Valve CK-ES3340 During PMT; Dated October 17, 2015
- CR-PLP-2015-05106; Boric Acid Buildup on MV-PC1085B, P-50D; Dated October 17, 2015
- CR-PLP-2015-05109; Relief Valves Lifting On Feedwater Heaters E-7A and E-4A On P-2B Condensate Pump Start; Dated October 17, 2015
- CR-PLP-2015-05110; Charging Loop 2A Stop Valve Has Minor Boric Acid Buildup On Its Packing Gland; Dated October 17, 2015
- CR-PLP-2015-05114; A Bank Of Heaters Is Not Energized on LCC-15; Dated October 17, 2015
- CR-PLP-2015-05115; Items Identified During the SOP-1A Checklist 1.4 Containment Closeout Walkthrough; Dated October 17, 2015
- CR-PLP-2015-05116; Dry White boric Acid Found on the Hingne Side Of MZ-19 Personnel Air Lock; Dated October 17, 2015
- CR-PLP-2015-05127; Upper Motor Bearing Service Water Cooling Outlet Piping Connected to P-2B Is leaking; Dated October 17, 2015
- CR-PLP-2015-05131; Possible Ground On CV-3040; Dated October 17, 2015
- CR-PLP-2015-05133; PCV-2071, Volume Control Tank Hydrogen Supply, Is Failed and Unable to Maintain pressure; Dated October 17, 2015
- CR-PLP-2015-05134; Received Alarm EK-0913, Primary Coolant Pump Vibration Alert Unexpectedly; Dated October 18, 2015
- CR-PLP-2015-05138; CV-0710, ‘B’ MFP Recirc Control Valve Indicates Open while CV-0711 ‘A’ MFP Recirc Control Valve Indicates Closed; Dated October 18, 2015
- CR-PLP-2015-05143; Gas Void with A 4 Inch Arc Measured At ABS Point #3; Dated October 18, 2015
- CR-PLP-2015-05144; Gas Void Not Found At ABS Point #20; Dated October 18, 2015
- CR-PLP-2015-05145; Significant Leak by on CV-30125 When Closed for Shutdown Cooling; Dated October 18, 2015

- CR-PLP-2015-05147; PMT For Replacement of Main Transformer Cooler Breakers (EC45528) Conducted Under WO 384767 Task 13; Dated October 18, 2015
- CR-PLP-2015-05149; Leakage From A Valve Manifold Connected To and Directly Above LT-0758B; Dated October 18, 2015
- CR-PLP-2015-05150; P-50B Primary Coolant Pump Seal Face Is Covered In Water; Dated October 18, 2015
- CR-PLP-2015-05151; Leak Found On Small Piping End Cap Of PCS system; Dated October 18, 2015
- CR-PLP-2015-05153; Reactor Coolant Pump On-line Baker Testing Not Performed Per 1R25 Outage Schedule; Dated October 18, 2015
- CR-PLP-2015-05155; Received Letdown Relief Valve High Temperature Alarm While Changing Charging/Letdown Alignment From Double to Single; Dated October 18, 2015
- CR-PLP-2015-05187; Alarm EK-0102 Did Not Alarm during Main Turbine Protective Trip Testing; Dated October 19, 2015
- CR-PLP-2015-05194; Bonnet Steam Leak On CV-0571B; Dated October 20, 2015
- CR-PLP-2015-05196; Primary Coolant System Chemistry Was Outside Of Industry Guidelines; Dated October 20, 2015
- CR-PLP-2015-05197; Quadrant Power Tilt Incore/Encore Agreement Is Not Within Specification Per MT-10; Dated October 20, 2015
- CR-PLP-2015-05213; Several Incore Signals Deviating Higher Than Expected During Power Ascension; Dated October 20, 2015
- CR-PLP-2015-05223; CV-0605 Has Packing Leakage; Dated October 20, 2015
- CR-PLP-2015-05224; Check Valve CK-N2/459 is Leaking By; Dated October 20, 2015
- CR-PLP-2015-05230; Live Load Washers Misaligned on LCV-0605; Dated October 21, 2015
- CR-PLP-2015-05262; Incore 32 Detector 1 (Bottom) INCE-51M16 Behaving Erratically; Dated October 22, 2015
- CR-PLP-2015-05254; Maintenance Groups are Not Consistently Meeting Requirements of EN-OM-123; Dated October 21, 2015
- CR-PLP-2015-05305; A Main Feed Pump (P-1A) Was Supplied From Both high Pressure and Low Pressure Steam Sources; Dated October 26, 2015
- CR-PLP-2015-05306; The Axial offset-Axial Shape Index Deviation Is Limiting Control Rods to <128"; Dated October 26, 2015
- CR-PLP-2015-05543; An 8000 Pound Load of Lead Blankets and Their Associated Trollies Were Dropped In Containment While Suspended From the L-1 Auxiliary Hook; Dated November 2, 2015
- CR-PLP-2015095207; Deviation Written for New Breaker 52-435 Main Transformer Cooling No. 1; Dated October 20, 2015
- CR-PLP-2105-04734; CV-0608, Moisture Separator Drain Tank T-5 Level control As Found Condition of Actuator Very Poor; Dated October 6, 2015
- CR-PLP-2-15-03974; Mode 3 Walkdown VHX-4 Containment Air Cooler; Dated September 16, 2015
- EA-EC60406-01; Evaluation Of As-Built T-82A/B/C/C Steel Bracket To Support Welded Connection; Revision 0
- EC-45528; EX-10, Transformer Cooling Breaker Replacements; Revision 0
- EC-60406; Evaluation of As-Built T-82A/B/C/D Bracket To support Teel Welded Connection; Revision 0
- EC-60691; EX-10, Transformer Cooling Breaker Replacement for 52-435 and 52-555; Revision 0
- EC-60783; EX-10, Replacement of Cooling Breakers 8-9 and 8-10; Revision 0
- ECN-60416; Revision to Breaker Curves for 52-435 and 52-444; Revision 0
- EN-DC-319; Boric Acid Corrosion Control Program (BACCP); Revision 11

- EN-IS-102; Confined space Program; Revision 8
- EN-MA-118; Foreign Material Exclusion; Revision 10
- EN-MA-119; Material Handling Program; Revision 223
- EN-OM-123; Fatigue Management Program; Revision 11
- EN-RE-326; PWR Core Loading Verification; Revision 1
- EN-RE-326; PWR Core Loading Verification; Revision 1
- EN-RE-327; PWR Startup Critical Predictions and Evaluation Process; Revision 3
- EOP Supplement 1; Pressure Temperature Limit Curves; Revision 5
- GOP-11; Refueling Operations and Fuel Handling; Revision 48
- GOP-2; Mode 5 to Mode 3 ≥525° F; Revision 39
- GOP-3; Mode 3 ≥525° F to Mode 2; Revision 31
- GOP-4; Mode 2 to Mode 1; Revision 23
- MSM-M-71; "Containment Cleanliness Implementation Plan and Containment Closeout," Revision 11
- MSM-M-72; Movement of Heavy loads In Turbine Building; Revision 1
- Notified By NRC Senior Resident Inspector Of 2 Pair Of Vendor Stock Issue Gray Work Gloves Were Identified In A Cable Train In Room 106; Dated October 14, 2015
- PO-2; "PCS Heatup/Cooldown Operations;" Revision 7
- RFL-R-16; "Reactor Vessel Closure Head Installation;" Revision 16
- RFL-V-7; "Fuel Movement;" Revision 14
- RFL-V-9; Core Mapping System Setup and Operation; Revision 7
- RFL-V-9; Core Mapping System Setup and Operation; Revision 7
- RT-191; "Startup Physics Test Program; Revision 10
- RT-92; "Inspection of Containment sump Envelope;" Revision 8
- SOP-1C; "Primary Coolant System - Heatup;" Revision 21
- SOP-24; Ventilation and Air Conditioning System; Revision 68
- SOP-27; Spent Fuel Pool Cooling System; Revision 69
- SOP-28; Fuel Handling System; Revision 53
- SOP-28; Fuel Handling System; Revision 53
- SOP-2A; "Chemical and Volume Control System;" Revision 85
- SOP-2B; Attachment 2, Checklist 2.1: CVC System Checklist; Revision 49
- SOP-6; "Reactor Control system;" Revision 35
- WI-MSM-M-29; Installation and Removal of Primary Coolant System Vacuum Refill Equipment; Revision 5
- WO 334982; EMA-2103, P-50A Primary Coolant Pump, Lower Oil Reservoir Low
- WO 342445; MV-CVC2225, Letdown Orifice Bypass Stop Valve, Clean, Inspect And Repair
- WO 366824; CRD, Perform Dye-Penetrant Testing on CRD Seal Housings
- WO 372446; CRD-29, Inspect and Repair leakage At Autoclave
- WO 374147; EC-47349 (EC#16) Install Flex Connection in EB-20
- WO 380851; N-50, Remove/Install 45 CRD Drive & Seal Packages
- WO 384767; 52-435, Replace Breaker per EC-45528
- WO 384770; 52-555, Replace Breaker per ED-45528
- WO 390189; CV-3038, Replace Actuator Diaphragm and Actuator
- WO 396142; Inspect Entire Length of Intake Pipe and Lakebed
- WO 404046; MO-3066, Leaks By
- WO 407277; CV-0601, Remove Temp Mod (ED-55839) Stop Block from Valve
- WO 426244; MO-1043A, Replace Motor Connection Lubs
- WO 427469; MO-3066, Valve Seat Found Unrepairable, Replace Valve
- WO 427588; CV-0601, Install Valve Block per EC55839
- WO 427590; CV-0605, Install Valve Block per EC55839
- WO 427685; LS-0160, Reactor Vessel Flange Leak Drain Level Switch

- WO 427687; SNB-52, Steam Generator ‘A’ Snubber Spring Assembly Oil Leak
- WO 427913; K-2, Main Turbine Turning Gear Won’t Disengage
- WO 427978; FS-0885, Won’t Clear Alarm EK-1347
- WO 428012; CV-0152, Nitrogen Leak Thru Packing
- WO 428014; PCV-2071, Hydrogen to VCT Is Either Stuck Or Needs Adjustment
- WO 428291; LCV-0605, Packing Leak
- WO 52326163; CV-0781, Disassemble and Inspect PM
- WO 52326164; CV-0781 Failed A Drop Test Per EN-MA-143; Dated September 24, 2015
- WO 52405441; P/S-0701, Replace AFAS Actuation Channel A Remote Switch Power Supply
- WO 52405441; P/S-0771, Replace AFAS Actuation Channel ‘A’ Remote Switch Power Supply
- WO 5254892; CV-2099, Perform Diagnostic Testing
- CR-PLP-2015-03957; Mode 3 Walkdown – Bio Shield Penetration; Dated September 16, 2015
- CR-PLP-2015-04167; A Work Hour Waiver is Required for Covered Workers in the Mechanical Maintenance Department; Dated September 23, 2015
- CR-PLP-2015-05254; Maintenance Groups are Not Consistently Meeting Requirements of EN-OM-123; Dated October 21, 2015

<u>1R21 Component Design Bases Inspection</u>

- 5-39-R11-391; Show Results of New Evaluation of CCW Vulnerability to HELB; Dated August 22, 1990
- 90-1063; FSAR Change Request Regarding CCW Vulnerability to a HELB Inside Containment; Dated July 27, 1990
- CR-PLP-2015-01872; Condition Report - NRC CDBI Questions Regarding Classification of the Component Cooling Water System; Dated May 16, 2015
- CR-PLP-2015-01872; Operability Evaluation – NRC CDBI Questions Regarding Classification of the Component Cooling Water System; Dated May 15, 2015
- CR-PLP-2015-01873; NRC Questions Regarding Classification of CVCS and Response to a Design Basis Earthquake; Dated May 6, 2015
- CR-PLP-2015-05468; NRC CDBI Preliminary Violations Regarding Classification of the CCW System; Dated November 3, 15
- D-PAL-89-061; Post-Accident Operation of CCW System; Dated June 26, 1990
- D-PAL-89-120; Loss of Instrument Air – CCW System; Dated April 25, 1990
- EA-GWO 7793-01; CCW Piping Inside Containment; Revision 0
- FC-452-2; Revise CCW to Containment Isolation Logic; Dated November 19, 1984
- FC-657; CCW Isolation Logic Modification; Dated March 11, 1987
- GIP-2; Generic Implementation Procedure for Seismic Verification of Nuclear Power Plant Equipment; Dated February 14, 1992
- GL 87-02; Supplement No. 1 to GL 87-02 that Transmits Supplemental Safety Evaluation Report No. 2 on SQUG GIP-2; Dated 1992
- LER 89-006-01; CCW Availability Following a HELB; Dated September 19, 1989
- Letter; Evaluation Report on SEP Topic VI-4, Containment Isolation System for the Palisades Nuclear Power Plant, Unit 1; Dated May 11, 1981
- Letter; Forwarding Final Evaluation Report of SEP Topic VI-4, Containment Isolation System for the Palisades Nuclear Power Plant; Dated February 8, 1982
- Letter; Palisades – Evaluation of SEP Topic VII-3, Systems Required for Safe Shutdown (EICS Matters); Dated December 31, 1981
- Letter; Palisades – SEP Topics V-10.B, RHR System Reliability, V-11.B, RHR Interlock Requirements and VII-3, Systems Required for Safe Shutdown (Safe Shutdown Systems Report); Dated December 23, 1981

- Letter; Palisades – SEP Topics V-10.B: RHR System Reliability, V-11.B, RHR Interlock Requirements and VII-3, Systems Needed for Safe Shutdown (Safe Shutdown Systems Report); Dated October 27, 1981
- Letter; Palisades Nuclear Plant – Final Closeout of USI A-46 Outliers; Dated June 26, 2003
- Letter; Palisades Nuclear Plant – SQUG Outlier Resolution – Revision of Commitment; Dated March 24, 2003
- Letter; Palisades Plant – Resolution of USI A-46, Verification of Seismic Qualification of Equipment in Operating Plants; Dated September 25, 1998
- Letter; SEP Safety Topics III-6, Seismic Design Consideration, and III-11, Component Integrity – Palisades Nuclear Power Plant; Dated December 18, 1981
- Letter; SEP Topic III-1, Quality Group Classification of Components and Systems (Palisades); Dated December 28, 1981
- Letter; SEP Topic III-1, VI-7.A.3, VI-7.B, VI-7.F, VI-10.A, VII-1.A and VII-2 (Palisades Plant); Dated November 21, 1980
- Letter; SEP Topic III-4.C, “Internally Generated Missiles” – Palisades; Dated September 21, 1981
- Letter; SEP Topic IX-3, Station Service and Cooling Water Systems Palisades; Dated February 8, 1982
- Letter; Status of SEP Topics III-1, III-4.B, III-5.A, IV-2, IX-1, and IX-4 (Palisades); Dated February 8, 1982
- NUREG-0820; Integrated Plant Safety Assessment – Systematic Evaluation Program – Palisades Plant; Dated October 1982
- Palisades FSAR; Revision 0
- Palisades FSAR; Revision 31
- Report; USI A-46 Equipment Evaluation Report – Palisades; Dated May 19, 1995
- SDR-03-1073; CVCS Declassification; Dated September 22, 2003
- SDR-99-0884; Cancellation of TS Surveillance Procedures and Revision of FSAR; Dated July 22, 1999

<u>1R22 Surveillance Testing</u>

- Admin 3.26; Implementation of Palisades Renewed License Requirements; Revision 3
- ARP-5; Primary Coolant Pump Steam Generator and Rod Drives Scheme, EK-09 (C-12); Revision 103
- Basis Document for RO-97; Auxiliary Feedwater System Automatic Initiation Test Procedure; Revision 8
- CR-PLP-2015-04597; Open 6 Inch ILRT Fill Line Full of Bird’s Nesting Material For At Least 10 Feet; Dated October 2, 2015
- CR-PLP-2015-04645; During RT-8C, Engineered Safeguards System, Acceptance Criteria Step 6.6, Left Channel, Was Not Initially Met; Dated October 4, 2015
- CR-PLP-2015-04758; Right Train SIS Block Relay “XB-R” Exhibited Evidence of Either High Contact Resistance or low Contact Pressure; Dated October 6, 2015
- CR-PLP-2015-04859; During Installation of ICI At Flange Location 2-5 The Installation Crew could Not Fully Insert the ICI Fully; Dated October 9, 2015
- CR-PLP-2015-04900; Core Exit Thermocouple 30 Is Reading Lower Than The Other Four CETs Currently Connected; Dated October 10, 2015
- CR-PLP-2015-04979; Primary System Drain Tank Read 16 psig at 0713 and 34 psig on October 13 Instead of 0 As Normal; Dated October 13, 2015
- CR-PLP-2015-04982; Primary Coolant Pump P-50A Oil Level Indication Sight Glass Appears to Have A Very Slow Leak; Date3d October 13, 2015

- CR-PLP-2015-04985; Post ILRT Wide Range Steam Generator Levels, LI-0757 A&B and LI-0758 A&B, Do Not Agree Within 20% As Required By MO-45 And Need Cal or Backfill; Dated October 13, 2015
- CR-PLP-2015-04998; RT-36 – IA Connection A Containment Wall PMT, Failed Due to Air Leakage; Dated October 14, 2015
- CR-PLP-2015-05016; Cables Need to Be Swapped On Incores; Dated October 14, 2015
- CR-PLP-2015-05231; Incore 26 And Incore 10 Bottom Detectors Have A Noisy Signal That Is Not Trending With Other Symmetric Incore Signals; Dated October 21, 2015
- CR-PLP-2015-05466; Neutron Flux Signal From the Top Rhodium Detector In Incore Core Instrument String 10 In Core Location J07 Has Degraded; Dated November 3, 2015
- CR-PLP-2015-06082; During The Performance of RO-97B Scheduled for December 15, 2015 Engineering Will Collect RPM and video Data on P-8B; Dated December 9, 2015
- CR-PLP-2015-06191; Auto Start Test Pump P-8A did Not Illuminate on C-11 As Expected; Dated December 15, 2015
- DBD-1.03; Auxiliary Feedwater System; Revision 9
- E-17; Containment High Pressure Sheet 6; Revision 11
- EN-DC-334; Primary Containment Leakage Rate Testing (Appendix J); Revision 3
- EN-OP-115; Conduct of Operations; Revision 015
- M-207; Auxiliary Feedwater System, Sheet 2; Revision 41
- M-215; Plant Heating System, Sheet 1; Revision 95
- QO-21; Basis Document for QO-21, Inservice Test Procedure – Auxiliary Feedwater Pumps; Revision 15
- QO-21; Inservice Test Procedure – Auxiliary Feedwater Pumps; Revision 46
- RO-19; Control Rod Position Verification; Revision 25
- RO-97; Auxiliary Feedwater System Automatic Initiation Test Procedure; Revision 22
- RT-191; Startup Physics Test Program; Revision 10
- RT-36; Basis Document For RT-36, Containment Integrated Leak Rate Test; Revision 14
- RT-36; Containment Integrated Leak Rate Test; Revision 21
- SEP-APJ-010; Palisades Primary Containment Leakage Rate Testing (Appendix J) Program; Revision 4
- SEP-RLP-PLP-001; Palisades Renewed License Program; Revision 0
- SOP-6; Reactor Control System; Revision 35
- WO 52543173 01; RO-19, Control Rod Position Verification
- WO 52563835; RO-97A – Auxiliary Feedwater System Automatic Initiation Test
- WO 52565431 01; RO-12, Containment High Pressure and Spray System Tests
- WO 52572821; RT-191 –Startup Physics Test Program
- WO 52650825 01; QO-21A – P-8A, IST Auxiliary Feedwater System

1EP4 Emergency Action Level and Emergency Plan Changes

- 10 CFR 50.54(q) Screening; EN-EP-801, “Emergency Response Organization;” Revision; Dated May 27, 2015
- 10 CFR 50.54(q); Screening, EI-3, “Communications and Notifications,” Revision; Dated December 3, 2014
- 10 CFR 50.54(q); Screening, EI-6, “Off-Site Dose Calculation and Recommendations for Protective Actions;” Revision; Dated February 9, 2015
- 10 CFR 50.54(q); Screening, EI-6.9, “Automated Dose Assessment Program,” Revision; Dated December 3, 2014
- 10 CFR 50.54(q); Screening, EN-EP-306, “Drills and Exercises,” Revision; Dated February 22, 2015

- 10 CFR 50.54(q); Screening, EN-EP-306, “Drills and Exercises,” Revision; Dated June 23, 2015
- 10 CFR 50.54(q); Screening, EN-EP-307, “Hostile Action Based Drills and Exercises,” Revision; Dated February 22, 2015
- 10 CFR 50.54(q); Screening, EN-EP-308, “Emergency Planning Critiques,” Revision; Dated February 22, 2015
- 10 CFR 50.54(q); Screening, EN-EP-313, “Off-site Dose Assessment Using the Unified RASCAL Interface,” Revision; Dated May 17, 2015
- 10 CFR 50.54(q); Screening, EN-EP-801, “Emergency Response Organization,” Revision; Dated February 12, 2015
- 10 CFR 50.54(q); Screening, EN-TQ-110, “Emergency Response Organization Training,” Revision; Dated February 13, 2015
- 10 CFR 50.54(q); Screening, EN-TQ-110-01, “Fleet E-Plan Training Course Summary,” Revision; Dated November 11, 2014
- 10 CFR 50.54(q); Screening, SEP, “Site Emergency Plan,” Revision; Dated April 30, 2015
- EI-3; Communications and Notifications; Revision 31
- EI-3; Communications and Notifications; Revision 32
- EI-6; Off-Site Dose Calculation and Recommendations for Protective Actions; Revision 13
- EI-6; Off-Site Dose Calculation and Recommendations for Protective Actions; Revision 14
- EI-6; Off-Site Dose Calculation and Recommendations for Protective Actions; Revision 15
- EI-6.9; Automated Dose Assessment Program; Revision 6
- EI-6.9; Automated Dose Assessment Program; Revision 7
- EN-EP-305; Emergency Planning, 10 CFR5 0.54(q), Review Program; Revision 3
- EN-EP-306; Drills and Exercises; Revision 5
- EN-EP-306; Drills and Exercises; Revision 6
- EN-EP-306; Drills and Exercises; Revision 7
- EN-EP-307; Hostile Action Based Drills and Exercises; Revision 2
- EN-EP-307; Hostile Action Based Drills and Exercises; Revision 3
- EN-EP-308; Emergency Planning Critiques; Revision 2
- EN-EP-308; Emergency Planning Critiques; Revision 3
- EN-EP-801; Emergency Response Organization; Revision 11
- EN-EP-801; Emergency Response Organization; Revision 12
- EN-TQ-110; Emergency Response Organization Training; Revision 11
- EN-TQ-110; Emergency Response Organization Training; Revision 12
- EN-TQ-110-01; Fleet E-Plan Training Course Summary; Revision 1
- EN-TQ-110-01; Fleet E-Plan Training Course Summary; Revision 2
- EP 601; Public Education and Information; Revisions 10 and 11
- Palisades Nuclear Plant On-Shift Staffing Analysis; Revision 2
- Palisades Power Plant 2013 Population Update Analysis; Dated September 24, 2013
- Palisades Power Plant 2014 Population Update Analysis; Dated September 23, 2014
- Palisades Power Plant Development of Evacuation Time Estimates; Dated August 2012
- SEP; Palisades Nuclear Plant Site Emergency Plan; Revision 26

4OA1 Performance Indicator Verification

- CR-PLP-2015-00584; Monitored Components in the MSPI Cooling Water System; Dated February 4, 2015
- EN-DC-205; Maintenance Rule Monitoring; Revision 5
- NRC Performance Indicator Technique / Data Sheet, Mitigating Systems Performance Indicator, Emergency AC Power (MS06); Dated October 2014 through September 2015

- NRC Performance Indicator Technique / Data Sheet; Mitigating Systems Performance Indicator, Cooling Water Support (MS10 CWS 1); Dated October 2014 through September 2015
- NRC Performance Indicator Technique / Data Sheet; Mitigating Systems Performance Indicator, Cooling Water Support (MS10 CWS 2); Dated October 2014 through September 2015
- Palisades MSPI Basis Document; Dated December 21, 2011

4OA2 Problem Identification and Resolution

- CR-PLP-2015-01181; During MO-7A-2 K-6B, Emergency Diesel Generator 1-2, Failed to Start; Dated March 18, 2015
- CR-PLP-2015-04139; DEH Turbine Control Panel Drop 200 Engineering Console Power supply Is cycling On and Off; Dated November 12, 2015
- CR-PLP-2015-04227; P-50C Missing O-Ring; Dated October 27, 2015
- CR-PLP-2015-04302; Steam Erosion Found at North East Corner of K-1-HP Horizontal Joint; Dated September 25, 2015
- CR-PLP-2015-04630; CV-2012 Letdown Intermediate pressure Control Failed To Stroke From the Control Room in Both Manual and Automatic; Dated October 3, 3015
- CR-PLP-2015-04652; Incorrectly Installed Rose Buds Discovered During 252-101 Breaker Install; Dated October 4, 2015
- CR-PLP-2015-04692; CRD-41 Seal Housing Catastrophic Failure During Pressure Testing; Dated October 5, 2015
- CR-PLP-2015-04732; CK-ES3166, Containment Sump Outlet Check to West Safeguards: Failed To fully Open During As-Left Testing; Dated October 6, 2015
- CR-PLP-2015-04831; Upright Sprinklers Replaced With Pendant Sprinklers, Need To Create Work Request to Replace Pendant Sprinklers with Upright Sprinklers; Dated October 8, 2015
- CR-PLP-2015-04884; Gross Air leakage on SV-5057B Steam Generator A Main Steam Isolation Valve Supply CV-0510; Dated October 9, 2015
- CR-PLP-2015-04951; Incorrect Reassembling of Valve During Installation of The Bracket For LVDT on CV-0570; Dated October 12, 2015
- CR-PLP-2015-04952; Incorrect Reassembling of Valve During Installation of The Bracket For LVDT on CV-0570; Dated October 12, 2015
- CR-PLP-2015-05022; Air Seal Around the Valve Stem on the Closing Side of the Actuator on CV-3057 is Blown Out and Leaking Severely; Dated October 15, 2015
- CR-PLP-2015-05219; CV-0569 Has A Class 5 Steam leak Where The Valve Shaft Extends Through The Mummy Case; Dated October 20, 2015
- CR-PLP-2015-05269; Leak At K-1-HP, High Pressure Turbine Horizontal Joint In An Area Where Weld Repair of the Horizontal Joint Was Performed During 1R24; Dated October 22, 2015
- CR-PLP-2015-05314; Generator End Right Side of K-1-HP, High Pressure Turbine Horizontal Joint Has A New Leak That did Not Exist Prior to 1R24; Dated October 26, 2015
- CR-PLP-2015-05384; Need Work Request to Remove the South West Corner of the Main Generator Skirting to Enable Investigation of a Potential Minor Hydrogen leak; Dated October 29, 2015
- CR-PLP-2015-05642; Workers Were Directed To Continue Installing Non-conforming Fire Sprinklers; Dated November 12, 2015
- CR-PLP-2015-05771; Cation Needed to Evaluate the Need For a Filter on the CRD Test Stand; Dated October 5, 2015
- CR-PLP-2015-05931; During the Destructive Analysis of ASM-2A, It Was Identified That the Stop Nut Pin Hardness Was Approximately 17.5 HRC; Dated December 1, 2015

- DBD-5.01; Diesel Engine and Auxiliary Systems; Revision 7
- Document Number 1000385-FA; Failure Analysis of Air Start Motor; Revision 0
- Document Number 5S027.1; Seismic Test Report for an Ingersoll Rand Air Start Motor; Dated December 14, 2005
- EACE; K-6B (Emergency Diesel Generator 1-2) Failure to Start; Revision 0
- EACE; K-6B (Emergency Diesel Generator 1-2) Failure to Start; Revision 2
- EN-LI-118; Cause Evaluation Process; Revision 22
- EN-MA-123; Identification and Trending of Rework; Revision 8
- EN-OE-100-02; Operating Experience Evaluations; Revision 1
- MO-7A-2; Emergency Diesel Generator 1-2; Revision 87
- MO-7A-2; Emergency Diesel Generator 1-2; Revision 87
- Station Re-work Review Board Meeting Minutes; Dated December 17, 2015
- Three Year Trend Review of Re-work Evaluations; Dated January 1, 2012 through October 27, 2015
- WO 0374202; CK-ES3166, Containment Sump Outlet Check Valve to West Safeguards, Replace O-Ring Cartridge and Lever Operator Shaft
- WO-0427396; Replace Pendent Sprinklers In Bus 1C With Upright Sprinklers

4OA3 Follow-Up of Events and Notices of Enforcement Discretion

- Admin 4.08; Post Event Review; Revision 7
- ARP-2; Generator Scheme EK-03 (EC-11); Revision 54
- CR-PLP-2015-3826; Plant Experienced a Turbine Trip/Reactor Trip Due to Loss of Power to DEH; Dated September 16, 2015
- CR-PLP-2015-3827; Turbine/Reactor Trip on Turbine Panel Trouble; Dated September 16, 2015
- EOP-1.0; Standard Post-Trip Actions; Revision 16
- EOP-2.0; Reactor Trip Recovery; Revision 13
- GOP-8; Power Reduction and Plant Shutdown to Mode 2 or Mode 3 >525°F; Revision 36
- Licensee Event Report 2015-001-00; Automatic Reactor Trip Results from a Turbine Trip Initiated from the Digital Electro-Hydraulic Control System; Dated November 10, 2015
- Operations Logs for September 15-16, 2015
- Post Event Review Report for Turbine Trip / Reactor Trip Due to Loss of Power to DEH on September 16, 2015

4OA5 Other Activities

- AOP-38; Acts of Nature Basis Document; Revision 5
- AOP-38; Acts of Nature; Revision 5
- Apparent Cause Evaluation; Adverse Trend in Human Performance (CR-PLP-2014-03898); Dated August 27, 2014; Revision 0.
- Calculation EA-EC55593-01; Beyond Design Basis (BDB) Evaluation: Local Intense Precipitation Flow Through Manhole 4 to Manhole 1; Revision 0
- CR-PLP=2015-00575; Inspect Steam Trap ST-0520 and the Associated Piping; Dated February 23, 2015
- CR-PLP-2011-5723; Auxiliary Feedwater Pump, P-8B, Overspeed Trip Actuation; Dated October 28, 2011
- CR-PLP-2013-3865; The Senior NRC Resident Identified That There Was Not a CR Written to Document Specifically That as a Result of the Heavy Rain Storm on 8/7/13 There Was Water Intrusion Into the Turbine Building; Dated August 20, 2013
- CR-PLP-2014-05477; ACE for TDAFW Overspeed Trip; Dated November 18, 2014

- CR-PLP-2014-05483; During the Performance of Testing P-8B, Auxiliary Feedwater Pump, per RO-145, Comprehensive Pump Test Procedure As Found Shaft Speed Was Outside Its Band; Dated November 15, 2014
- CR-PLP-2014-05822; Job aid contains incorrect information; Dated December 14, 2014
- CR-PLP-2015-00393; Monitor Steam Pressure to K-8 Turbine Monitor and Record Turbine RPM; Dated January 23, 2015
- CR-PLP-2015-00199; Worker reluctance to identify an issue; Dated January 13, 2015
- CR-PLP-2015-00227; Staffing levels in security; Dated January 14, 2015
- CR-PLP-2015-00616; Supervisor behavior below standards; Dated February 8, 2015
- CR-PLP-2015-00964; ST-0523, K-8 Exhaust Steam Trap to Drain, Found Several Pieces of Carbon Graphite FME in The Bottom Of The Bowl; Dated March 3, 2015
- CR-PLP-2015-00965; While Rebuilding ST-0523, K-8 Exhaust Steam Trap to Drain Replacement Steam Trap Bucket Provided Under CAT ID 2962910 Was Not Physically the Same As The bucket Installed in the Trap; Dated March 3, 2015
- CR-PLP-2015-00969; Found Stem and Disc of St-0522B to Be Severely Degraded; Dated March 3, 2015
- CR-PLP-2015-00971; A Facility Change (FC-966) Was Not Adequately Incorporated Into work Order 40435-01, "ST-0520, Inspect Steam Trap and In/Out Piping for Blockage;" Dated March 4, 2015
- CR-PLP-2015-00977; K-8 Steam Supply Line Drain Has A 50 DPM Leak; Dated March 4, 2015
- CR-PLP-2015-01040; Install the TDAFW New Resetting lever and latch Plate with Raked Edges; Dated March 10, 2015
- CR-PLP-2015-01871; Staffing levels in security; Dated May 6, 2015
- CR-PLP-2015-01935; Ascertain the Configuration of the Steam Supply To/From the Turbine Driven Aux Feed Water Pump Steam Pressure Control Valve CV-0522B; Date May 11, 2015
- CR-PLP-2015-02156; Staffing levels in security; Dated May 25, 2015
- CR-PLP-2015-02514; Security key organization; Dated May 25, 2015
- CR-PLP-2015-04244; Undue pressure on workers; Dated September 25, 2015
- CR-PLP-2015-05960; During A System Engineering Walkdown of Auxiliary Feedwater system, It Was Observed the Steam Trap ST-0520 Was Discharging Condensate Into the AFW Pump Room Drain; Dated December 2, 2015
- CR-PLP-2015-0784; A Work Request is Needed to Seal the 10 Conduits Leaving the West Side of Manhole 4 (MH-4); Dated February 19, 2015
- CR-PLP-2015-5730; The Flood Hazard Reevaluation Report Did Not Identify Potential Interim Actions; Dated November 19, 2015
- CR-PLP-2015-6103; Walkdowns to Determine Sandbag Configuration for CR-PLP-2015-05730 Determined that Door-107 Did Not Include Signage to Block Access; Dated December 10, 2015
- CR-PLP-2015-6155; While Reviewing Report PLP-RPT-15-00010 Found Two Editorial Errors; Dated December 14, 2015
- CR-PLP-2015-6160; While Reviewing Report PLP-RPT-15-00010 Found an Editorial Error; Dated December 14, 2015
- DBD-1.03; Auxiliary Feedwater system; Revision 8
- Drawing E-301; Sheet 2, Ductbanks and Underground Electric Lines Between Plant and Switchyard; Revision 9
- EA-FC-966-01; Mechanical Design Basis Analysis for AFW Steam Supply; Revision 3
- EC 55519; K-8 (Auxiliary Feed Pump P-8B Turbine Driver): Replace the Standard Knife Edge Arrangement for the Overspeed Trip Mechanism with Those with Raked Edges As Recommended By the Vendor; Revision 1

- EC-55520; Overspeed Trip Setting Increase and Trip Pin to Plunger Clearance Adjustment of the Steam Driven Auxiliary Feedwater Pump (P-8B) Turbine Driver K-8 In Accordance With Manufacturer Recommended Values; Revision 0
- Employee Concerns Program Listings of Cases, 2014, 2015
- EN-DC-153; Preventative Maintenance Component Classification; Revision 12
- EN-DC-204; Maintenance Rule Scope and Basis; Revision 3
- EN-DC-205; Maintenance Rule Monitoring; Revision 5
- EN-DC-324; Preventive Maintenance Program; Revision 15
- EN-LI-121; Trending and Performing Review Process; Revision 17
- EN-LI-121; Trending and Performing Review Process; Revision 18
- EN-QV-134; Employee Survey Response Protocol; Revision 2
- EN-QY-136; Nuclear Safety Culture Monitoring; Revision 5
- EN-QY-136; Nuclear Safety Culture Monitoring; Revision 6
- Entergy Organization Health Index March 2015 Survey Results; no date
- FLP-SOER-SOER10-2 PLP SECQUAL; Palisades Non-Qualified Security Event; no date.
- Letter to Anthony J. Vitale from the NRC; Palisades Nuclear Plant – Staff Assessment of the flooding Walkdown Report Supporting Implementation of Near-Term Task Force Recommendation 2.3 Related to the Fukushima Dai-ichi Nuclear Power Plant Accident; Dated June 17, 2014
- Letter to the NRC from Anthony J. Vitale, Required Response 2 for Near-Term Task Force Recommendation 2.1: Flooding - Hazard Re-Evaluation Report; Dated March 11, 2015
- Letter to the NRC from Charles Arnone; Flooding Walkdown Report – Response to NRC Request for Information Pursuant to 10 CFR 50.54(f) Regarding the Flooding Aspectos of Recommendation 2.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident; Dated November 27, 2012
- Letter to Vice President; Operations at Palisades Nuclear Plant from Victor E. Hall, NRC/NRR, Palisades Nuclear Plant – Interim Staff Response to Reevaluated flood Hazards Submitted In Response To 10 CFR 50.54(f) Information Request – Flood-Causing Mechanism Reevaluation (:AC No. MF6128); Dated December 23, 2015
- MSM-M-16; Inspection of Watertight Barriers; Revision 17
- Night & Standing Order Log; Dated December 12, 2015
- Nuclear Safety Culture Assessment; Entergy Nuclear; Dated June-July 2015, November 18, 2015, (portions of report considered SYNERGY Proprietary)
- Palisades Listing of Allegations – January 1, 2014 to November 2015, October 14, 2015
- Palisades Problem Identification and Resolution Focused Inspection Report 05000255/2014009; Dated March 6, 2014
- Palisades Problem Identification and Resolution Inspection Report 05000255/2014007; Dated June 20, 2014
- Palisades Safety Conscious Work Environment Issue of Concern Follow-Up Inspection Report 05000255/2014011; Dated January 20, 2015
- Palisades Supplemental Inspection Report 05000255/2012011; Dated November 9; 2012
- PNP 2014-069; Energy Nuclear Operations; Reply to Inspection Report 05000255/2014007; Dated July 18, 2014
- PNP 2015-042; Entergy Nuclear Operations; Inc. One-Year Status Notification in Response to Confirmatory Order, EA-14-013; Dated June 19, 2015
- QO-21; Inservice Test Procedure – auxiliary Feedwater Pumps; Revision 46
- Safety Culture Survey; Engineering, Nuclear Oversight, Maintenance; Operations, Radiation Protection; Dated March 2015
- Safety Culture Survey; Performance Improvement; Emergency Planning, Regulatory Assurance and Security Departments; Dated January 2015
- Security Leader Development Program syllabus and agenda; no date

- SEP-PDM-PLP-001; Palisades Ultrasonic Monitoring Program Section; Revision 0
- Synergy 2015 Nuclear Safety Culture Assessment June – July 2015 preliminary results; Dated August 4, 2015
- Synergy 2015 Nuclear Safety Culture Assessment June – July 2015, September 9, 2015
- USNRC; Confirmatory Order Related to NRC Report No. 05000255/2014406 and OI Report 3-2013-018, Palisades Nuclear Plant; Dated July 21, 2014
- VEN-M101-3222; Stress isometric for Steam to P-8B Turbine, Sheet 7; Revision 4
- VEN-M101-3223; Stress Isometric for Steam to P-8B Turbine, Sheet 8; Revision 4
- WO 00401961; ST-0512, Inspect Trap Intermals & Piping for Blockage
- WO 00401963; ST-0513, Inspect Trap Internals & Piping for Blockage
- WO 00401965; ST-0523, Inspect Trap Internals & Piping for Blockage
- WO 00404535; ST-0520, Inspect Steam Trap and In/Out Piping for Blockage
- WO 398228; P-8B/K-8, AFW Pump, Tripped on Overspeed
- WO 406801-01; MH-4, Seal 10 Conduits Leaving West Side of Manhole
- WO 408644; K-8B, Install New Reset Lever and latch Plate (scheduled 2/10-2/11/16)
- WO 413732; CV-0522B, Verify Configuration of Steam Supply Line
- WO 52635952; Annual Inspection of Watertight Barriers
- WO 5264828; QO-21B, P-8B IST Auxiliary Feedwater System
- WO ST-0522B; Inspect Trap Internals & Piping for Blockage
- WT-WTPLP-2015-0058; Engineering Actions to Track for 2015; Dated April 10, 2015

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents and Access Management System
AFW	Auxiliary Feedwater
ANSI	American National Standards Institute
AOV	Air Operated Valve
ASME	American Society for Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BMV	Mare Metal Visual
CAP	Corrective Action Program
CDBI	Component Design Bases Inspection
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CHP	Containment High Pressure
CIV	Containment Isolation Valve
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBE	Design Basis Event
DEH	Digital Electro-Hydraulic
DG	Diesel Generator
DPU	Distributed Processing Units
EACE	Equipment Apparent Cause Evaluation
EAL	Emergency Action Level
ECP	Employee Concerns Program
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ET	Eddy Current Testing
GDC	General Design Criteria
GIP	Generic Implementation Procedure
HELB	High Energy Line Break
ICI	Incore Instrumentation
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
LER	Licensee Event Report
MSPI	Mitigating Systems Performance Indicator
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
OPC	Overspeed Protection Control
ORAT	Outage Risk Assessment
PARS	Publicly Available Records System
PCP	Primary Coolant Pump
PCS	Primary Coolant System
PI	Performance Indicator
PM	Preventive Maintenance
PT	Dye Penetrant
QA	Quality Assurance

RFO	Refueling Outage
RWP	Radiation Work Permit
SDP	Significance Determination Process
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SFP	Spent fuel Pool
SG	Steam Generator
SIS	Safety Injection Signal
SL	Severity Level
SQUG	Seismic Qualification Utility Group
SRRB	Station Rework Review Board
SSE	Safe Shutdown Earthquake
SSC	Structures; Systems; and Components
SWS	Service Water System
TI	Temporary Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
USI	Unresolved Safety Issue
UT	Ultrasonic
WO	Work Order

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Sincerely,

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