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

May 7, 2014

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

SUBJECT: PALISADES NUCLEAR PLANT INTEGRATED INSPECTION REPORT 05000255/2014002

Dear Mr. Vitale:

On March 31, 2014, 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 April 11, 2014, with you and other members of your staff.

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

Based on the results of this inspection, two NRC-identified and three self-revealed findings of very low safety significance were identified. Four of the findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the violations as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

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

If you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Palisades Nuclear Plant.

Additionally, as we informed you in the most recent NRC integrated inspection report, cross-cutting aspects identified in the last 6 months of 2013 using the previous terminology were being converted in accordance with the cross-reference in Inspection Manual Chapter (IMC) 0310. Section 4OA5 of the enclosed report documents the conversion of these cross-cutting aspects, which will be evaluated for cross-cutting themes and potential substantive cross-cutting issues in accordance with IMC 0305 starting with the 2014 mid-cycle assessment review. If you disagree with the cross-cutting aspect assigned, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Palisades Nuclear Plant.

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

Sincerely,

/RA/

Eric Duncan, Chief
Branch 3
Division of Reactor Projects

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

Enclosure:
Inspection Report 05000255/2014002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	05000255/2014002
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	January 1 through March 31, 2014
Inspectors:	T. Taylor, Senior Resident Inspector A. Garmoe, Senior Resident Inspector A. Scarbeary, Resident Inspector T. Bilik, Reactor Engineer J. Cassidy, Senior Health Physicist G. Hansen, Security Inspector M. Jones, Reactor Inspector J. Lennartz, Project Engineer M. Phalen, Senior Health Physicist E. Sanchez-Santiago, Reactor Inspector
Approved by:	Eric Duncan, Chief Branch 3 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000255/2014002, 01/01/2014 – 03/31/2014; Palisades Nuclear Plant; Equipment Alignment; Inservice Inspection Activities; Refueling and Other Outage Activities; Radiological Hazard Assessment and Exposure Controls.

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Five Green findings were identified by the inspectors or were self-revealed. Four of these findings were considered non-cited violations (NCVs) of NRC regulations. The significance of inspection findings are 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," dated June 2, 2011. Cross-cutting aspects were determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated January 1, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated July 9, 2013. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 5, dated February 2014.

A. <u>NRC-Identified and Self-Revealed Findings</u>

Cornerstone: Initiating Events

- <u>Green</u>. A finding of very low safety significance and an associated non-citied violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when licensee personnel failed to have an adequate procedure and work order (WO) to install steam generator nozzle dams. The licensee entered this issue in their Corrective Action Program (CAP) as Condition Report (CR) PLP-2014-00770, Improper Routing of Nozzle Dam Air Supply. As part of their corrective actions, the licensee planned to revise the nozzle dam installation procedure and the WO.

  The inspectors determined that this finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the finding was associated with the Procedure Quality attribute of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations, and was similar to the more than minor criteria in Example 5.a of IMC 0612, Appendix E, "Examples of Minor Issues." As it related to this finding, the intended design of the nozzle dam air supply system was not correctly translated into the installation procedure or the work instructions. Further, the nozzle dam air system was not properly tested prior to being placed into service. Since the plant was shutdown in Mode 6, the inspectors assessed the risk significance of the event in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process." A Phase 2 risk evaluation was required that determined the total event risk was 3.6E-8 and was therefore of very low safety significance (Green). This finding had an associated cross-cutting aspect in the Change Management (H.3) component of the Human Performance cross-cutting area. In particular, issues during the previous refueling outage led the steam generator project management team to review the configuration of the nozzle dam air system. Through this review, the licensee identified that changes to the alignment of air to the nozzle dams was required. However, due to turnover within the project management group and inadequate communications and documentation, the licensee failed to appropriately evaluate and implement those changes. (Section 1R04)

- Green. The inspectors identified a finding of very low safety significance and an associated non-citied violation of 10 CFR 50.55a(g)(6)(ii)(F)(3) when licensee personnel failed to complete required baseline volumetric examinations for nine dissimilar metal (DM) butt welds in the Primary Coolant System (PCS) that were fabricated from Inconel Alloy 82/182 weld metal and were susceptible to primary water stress corrosion cracking (PWSCC). The licensee entered this issue into their CAP as CR-PLP-2014-01742, NRC Question on Whether Hot and Cold Leg Branch Connection Welds are In Scope of ASME [American Society of Mechanical Engineers] Code Case (CC) N-770-1. As part of their corrective actions, the licensee submitted a request for relief to the NRC to allow substitution of a visual and dye penetrant surface examination of these welds as an alternative to volumetric examinations. The NRC granted verbal relief on March 13, 2014, which stated the licensee could implement the proposed alternative to 10 CFR 50.55a(g)(6)(ii)(F), which included a commitment to perform enhanced leakage monitoring during the current operating cycle and perform the required volumetric examinations during the next refueling outage.

  The inspectors determined that this finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” because the finding was associated with the Equipment Performance (Reliability) attribute of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors also determined that if left uncorrected the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, the failure to complete volumetric examinations on the nine DM butt welded PCS branch connections fabricated with Alloy 82/182 weld metal could have allowed PWSCC susceptible material to remain in service, which could propagate and result in a Loss-of-Coolant-Accident (LOCA). The inspectors performed a Phase I Significance Determination Process screening using IMC 0609, Appendix A, “The Significance Determination Process (SDP) for Findings At-Power,” Exhibit 1, “Initiating Events Screening Questions.” The inspectors answered the Phase I SDP “LOCA Initiators” Questions A1 and A2 ‘No’ because undetected cracks, if present, were not yet through-wall and did not challenge the structural integrity of the welds. Therefore, this finding screened as having very low safety significance (Green). This finding had an associated cross-cutting aspect in the Evaluation (P.2) component of the Problem Identification and Resolution cross-cutting area because the licensee did not ensure that the resolution of the issue appropriately addressed causes and the extent of condition. Specifically, when determining the applicability of CC N-770-1, the licensee failed to thoroughly evaluate the scope of welds susceptible to PWSCC that required volumetric examination commensurate with the safety significance of this issue. (Section 1R08.5 b)

- Green. A finding of very low safety significance and an associated non-citied violation of Technical Specification (TS) 5.4.1, “Procedures,” was identified by the inspectors when licensee personnel failed to follow procedure EN-MA-118, “Foreign Material Exclusion (FME),” during work on the safety-related critical service water (SW) system during refueling outage (RFO) 1R23. Specifically, Sections 5.2[1] and 5.2[6] of EN-MA-118 stated that planners and procedure writers should evaluate FME considerations for work activities and include job-specific FME controls in work instructions and procedures. Additionally, EN-MA-188 stated that during the planning stage, the planner should designate the FME Zone type, risk level, pathways to FME sensitive equipment, and work practice restrictions, as applicable, in all work packages. However, adequate controls were not established and documented when the decision was made to use an inflatable bladder inside the SW system when work was being performed on the system. As a result, on two separate occasions during RFO 1R23, bladders were inadvertently entrained into the return header of the SW system by the relative vacuum created by system flow. The licensee entered this issue into their CAP as CR-PLP-2014-00715, Vacuum was So Great that Bladder was Ripped Off Lanyard and Lost in Piping, and CR-PLP-2014-01176, FME Bladder Lost During Work Near CV-0823. As part of their corrective actions, the licensee successfully completed a comprehensive SW system test, which validated acceptable system parameters.

The inspectors determined that this finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. In accordance with Checklist 3, "PWR [Pressurized Water Reactor] Cold Shutdown and Refueling Operation RCS [Reactor Coolant System] Open and Refueling Cavity Level < 23' Or RCS Closed and No Inventory in Pressurizer Time to Boiling < 2 hours," following the loss of the first bladder, and Checklist 4, "PWR Refueling Operation: RCS Level > 23' Or PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer," following the loss of the second bladder of Attachment 1, "Phase 1 Operational Checklists for both PWRs and BWRs [Boiling Water Reactors]," of IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," the inspectors determined that mitigation capabilities were not adversely impacted. Additionally, utilizing Table 1, "Losses of Control," of IMC 0609, Appendix G, the inspectors determined there was no loss of control. As a result, the finding screened as having very low safety significance (Green). This finding had an associated cross-cutting aspect in the Work Management (H.5) component of the Human Performance cross-cutting area because the licensee did not implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority. In particular, the work process did not include the identification and management of risk commensurate to the work and the need for coordination with different groups or job activities. (1R20)

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance and an associated non-citied violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when licensee personnel failed to follow maintenance procedure RFL-R-16, "Reactor Vessel Closure Head Installation." Specifically, during the reactor vessel head lift on March 5, 2014, to support reinstallation onto the vessel flange, workers failed to identify an interference with the reactor head lift structure, causing the head to impact a jack screw on the structure and increasing the total load weight to approximately 283,000 pounds, which was greater than the procedural maximum polar crane load rating of 270,000 pounds. The licensee entered this issue into their CAP as CR-PLP-2014-01903, Reactor Head Flange Contacted Jacking Screw While Raising it Off the Head Stand. As part of their corrective actions, the licensee conducted a Level 1 Human Performance Evaluation, generated a site-wide Human Performance error communication, and performed work crew stand downs to discuss crane and rigging expectations.

  The inspectors determined that this finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the finding was associated with the Human Performance attribute of the Barrier Integrity cornerstone and adversely impacted the cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Additionally, the inspectors determined that the performance deficiency could reasonably be viewed as a precursor to a significant event and that if left uncorrected the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, the operability of the containment polar crane was required to be evaluated and the reactor vessel head was required to be inspected after the event occurred to verify no significant damage was caused and the maximum design limit of the crane could have been exceeded if the evolution was not stopped when it was, which increased the risk of dropping the head during the lift. The finding was screened in accordance with IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1, "Phase 1 Operational Checklists for both PWRs and BWRs." The finding was determined to be of very low safety significance (Green) based on not requiring a quantitative assessment after reviewing the five shutdown safety functional areas in Checklist 3, "PWR Cold Shutdown and Refueling Operation RCS Open and Refueling Cavity Level < 23' Or RCS Closed and No Inventory in Pressurizer Time to Boiling <2 hours." This finding had an associated cross-cutting aspect in the Challenge the Unknown (H.11) component of the Human Performance cross-cutting area. Specifically, human performance investigations identified that workers exhibited a lack of rigor when performing interference verifications prior to and during the reactor head lift, and an inadequate "stop when unsure" mentality when assessing the situation before continuing with the head lift. In addition, the workers and supervisors for this task did not understand that the load cell increase exceeded the procedural maximum value and did not inform decision-makers outside of the immediate work area to validate it was safe to proceed with the evolution. (Section 1R20)

Cornerstone: Occupational Radiation Safety

- Green. A finding of very low safety significance was self-revealed when workers received unplanned and unintended occupational radiation dose during a maintenance outage conducted in August 2012 due to deficiencies in the licensee's Radiological Work Planning and Work Execution Program. Specifically, the licensee failed to properly incorporate As-Low-As-Reasonably-Achievable (ALARA) strategies and insights while planning and executing Control Rod Drive Mechanism (CRDM) 24 housing work. The licensee entered this issue into their CAP as CR-PLP-2014-05812, UT [Ultrasonic Testing] Exams of the Additional CRDM Stalk Housings Has Exceeded the Dose Estimate for the RWP [Radiation Work Permit]. Corrective actions were implemented to address the outage planning and work execution issues.

  The inspectors determined that this finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the finding was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone and adversely impacted the cornerstone objective of ensuring the adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Additionally, the finding was similar to the more than minor criteria in Example 6.i of IMC 0612, Appendix E, "Examples of Minor Issues." The inspectors screened this finding in accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The inspectors determined that the finding did not involve: (1) a radiological overexposure; (2) a substantial potential for an overexposure; or (3) a compromised ability to assess dose.

The inspectors also determined that the finding involved ALARA planning and work controls and that the licensee’s 3-year rolling collective dose average was above 135 person-Rem at the time the performance deficiency occurred. However, because the work activity was a single occurrence that involved an actual dose outcome that was within the licensee’s control of less than 25 person-Rem, this finding was determined to be of very low safety significance (Green). This finding had an associated cross-cutting aspect in the Work Management (H.5) component of Human Performance cross-cutting area because the licensee did not plan work activities that appropriately incorporated radiological safety. (Section 2RS2)

REPORT DETAILS

Summary of Plant Status

The reactor operated at or near full power until January 19, 2014, when the plant was shut down for planned refueling outage (RFO) 1R23. On March 15, the reactor was taken critical and the plant was subsequently sychronized to the grid on March 16. The reactor achieved full power on March 18 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

1R04 Equipment Alignment (71111.04)

a. Inspection Scope

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

- ‘A’ High Pressure Safety Injection (HPSI) Train During ‘B’ HPSI Train Surveillance;
- Steam Generator Nozzle Dam Air Supply During RFO 1R23;
- ‘B’ Shutdown Cooling Train During RFO 1R23;
- Critical Service Water (SW) System Alignment for Component Cooling Water (CCW) Heat Exchanger Isolation; and
- SW System During Testing of Opposite Train.

The inspectors selected these systems based upon 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, the Updated Final Safety Analysis Report (UFSAR), Technical Specification (TS) requirements, outstanding work orders (WOs), condition reports (CRs), 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 corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

These activities constituted five partial system walkdown samples as defined in Inspection Procedure (IP) 71111.04-05.

b. Findings

Inadequate Installation of Steam Generator Nozzle Dams

Introduction: A finding of very low safety significance (Green) and an associated non-cited violation (NCV) of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when licensee personnel failed to maintain an adequate procedure and WO to install steam generator nozzle dams during RFO 1R23.

Description: On January 28, 2014, steam generator personnel identified steam generator nozzle dam low pressure alarms. An investigation revealed that the inlet air pressure in the system that supplied pressurized air to maintain the nozzle dams inflated had decreased from the nominal value of approximately 105 pounds per square inch gauge (psig) to approximately 20 psig. Subsequent licensee walkdowns and troubleshooting identified the following issues with the steam generator nozzle dam air system:

- A valve in the steam generator nozzle dam air system was identified to be closed when it should have been open. After opening this valve, system air pressure returned to normal. The cause or duration of the mispositioned valve could not be determined.

- Temporary air compressors that supplied air to the nozzle dams utilized hoses that were improperly routed through the containment hatch. Maintenance procedure RFL-SG-2, "Steam Generator Primary Nozzle Dam Installation and Removal," provided vague instructions on how to connect the air supply lines. Step 5.3.1.f of RFL-SG-2 stated, "connect air supply to control console and check for leakage under pressure."

- The backup air bottle air regulators did not properly maintain system air pressure after the primary air supply from the temporary air compressors was isolated when the isolation valve was closed. In accordance with RFL-SG-2, these regulators were procedurally required to be set to 40 psig. RFL-SG-2 included steps to verify that the regulators were set to 40 psig and those steps were marked to indicate that they had been performed. However, the nozzle dam air system low pressure alarm was received at about 38 psig and the lowest air pressure observed was approximately 20 psig, which was much lower than to 40 psig setpoint specified in RFL-SG-2.

The licensee completed an apparent cause evaluation (ACE) for the event. The identified apparent cause was that inadequate project management skills led to insufficient details in the procedures, inadequate communications, inadequate verifications, and a lack of interface with other groups (i.e. Operations). Contributing to the identified apparent cause was a lack of clear guidance on how to properly align the system for operation and an inadequate verification of the alignment.

The licensee entered this issue into their CAP as CR-PLP-2014-00770, Improper Routing of Nozzle Dam Air Supply. As part of their immediate corrective actions, the licensee re-opened the mispositioned valve to restore nozzle dam pressure. The backup air bottle regulators were also adjusted to control at the desire setpoint and the air hoses were properly routed. As part of their long-term corrective actions, the licensee planned to add details to RFL-SG-2 and the associated WOs to ensure proper air system alignment.

Analysis: The inspectors determined that the inadequate RFL-SG-2 procedure and WO to install the steam generator nozzle dams during RFO 1R23 was a performance deficiency that warranted a significance evaluation.

The inspectors determined that the finding was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," because it was similar to Example 5.a of IMC 0612, Appendix E, “Examples of Minor Issues.” This example described a design that was not correctly translated into work instructions and drawings and that would be a more than minor issue if the system was returned to service with that deficiency. In this case, the intended design of the nozzle dam air system was not correctly translated into the installation procedure and the work instructions. Further, the nozzle dam air system was placed in service with the aforementioned deficiencies and was not properly tested prior to being placed into service. This finding was also associated with the Procedure Quality attribute of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

Since the plant was shutdown in Mode 6, the inspectors assessed the risk significance of the event in accordance with IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process.” The inspectors reviewed Attachment 1, “Phase 1 Operational Checklists for Both PWRs [Pressurized Water Reactors] and BWRs [Boiling Water Reactors].” Considering the plant conditions that existed at the time of the event, the inspectors utilized Checklist 3, “PWR Cold Shutdown and Refueling Operation RCS [Reactor Coolant System] Open and Refueling Cavity Level < 23' Or RCS Closed and No Inventory in Pressurizer Time to Boiling < 2 hours.” The applicable line item in Checklist 3 was as follows:

- II.B.(3) - Training, procedures, and administrative controls implemented to avoid operations that could lead to perturbations in RCS level control or DHR [Decay Heat Removal] flow.

Therefore, Phase 1 criteria were met and the risk evaluation progressed to Phase 2. The Phase 2 risk evaluation was performed by a Region III Senior Reactor Analyst (SRA). The SRA reviewed IMC 0609, Appendix G, Attachment 2, “Phase 2 Significance Determination Process Template for PWR During Shutdown.” Given the plant conditions that existed at the time, the Plant Operating State was “POS-2.” The time window was "early" (TW-E) indicating that refueling had not yet been completed and decay heat was relatively high. The applicable initiating event was the Loss of Level Control (LOLC) initiating event.

For the LOLC initiating event frequency, the SRA used Table 1, “Initiating Event Likelihood (IEL) for LOLC Precursors.” Since there was a functioning check valve preventing leakage into the common collection system through the nozzle dams, the SRA assumed that more than 2 hours were available until the loss of decay heat removal function could have occurred after failure of the air supply. Also, the SRA credited the presence of accurate RCS level indication and that licensee actions to identify and recover the decay heat removal function if it were to be lost could have been readily performed. As a result, the IEL was “4” (i.e., 1E-04/year).

The mitigating functions for this initiator were evaluated using Worksheet 2, “SDP for a PWR Plant - Loss of Level Control in POS 2 (RCS Vented)," and Figure 6, "Event Tree for Loss of Level Control - POS-2." In Figure 6, two sequences were shown on the event tree ending in core damage. One sequence involved recovery of the decay heat removal function before depletion of water in the Safety Injection Refueling Water Tank (SIRWT) (i.e., “RHR-R”) and makeup to the SIRWT before its depletion and core damage (i.e., “RWSTMU”). The SRA assumed SIRWT depletion time to be more than 10 hours before core damage given the performance deficiency, and thus assigned a combined “5” for the mitigating functions in this sequence.

The remaining core damage sequence involved PCS injection before core damage (i.e., “FEED”). Given that there were multiple injection sources available, including both low pressure safety injection (LPSI) pumps, both charging pumps, and at least one HPSI pump, the SRA assigned the maximum allowable credit of “4” for the mitigating function in this sequence.

The total risk result of the internal event analysis is the sum of the individual results from the initiators above adjusted by the counting rule (i.e., multiply by 3.3) that is described in IMC 0609, Appendix A. The total internal event risk was subsequently calculated to be 3.6E-8. Therefore, the finding was determined to be of very low safety significance (Green).

The finding had an associated cross-cutting aspect in the Change Management (H.3) component of the Human Performance cross-cutting area. In particular, issues during the previous refueling outage led the steam generator project management team to review the alignment of the nozzle dam air system. Through this review, it was identified that changes were required for the alignment of air to the nozzle dams, however due to turnover within the project management group and inadequate communications and documentation, those changes were not properly evaluated and implemented.

<u>Enforcement</u>: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be accomplished in accordance with instructions, procedures, and drawings of a type appropriate to the circumstances.

Contrary to this requirement, procedure RFL-SG-2, “Steam Generator Primary Nozzle Dam Installation and Removal,” provided vague instructions on how to connect the air supply lines and the post-maintenance test in the associated WO simply stated to check for leakage once the air supply system was placed in service. This was revealed on January 28, 2014, when a valve within the air supply system was mispositioned and the back-up air supply bottles did not maintain system pressure at the expected value.

As part of their immediate corrective actions, the licensee re-opened the mispositioned valve to restore nozzle dam pressure. The backup air bottle regulators were also adjusted to control at the desire setpoint and the air hoses were properly routed. As part of their long-term corrective actions, the licensee planned to add details to RFL-SG-2 and the associated WOs to ensure proper air system alignment.

Because this violation was of very low safety significance and because the issue was entered into the licensee's CAP as CR-PLP-2014-00770, Improper Routing of Nozzle Dam Air Supply, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000255/2014002-01, Inadequate Installation of Steam Generator Nozzle Dams)**

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

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

- Fire Area 22: Turbine Lube Oil Room/Elevation 590' Turbine Building;
- Fire Area 13: 590' Elevation Auxiliary Building – General Areas;
- Fire Areas 2 and 3: Cable Spreading Room and 1-D Switchgear; and
- Fire Area 23: Turbine Building General Areas/Elevation 590', 607', 612', and 625'.

The inspectors reviewed these areas and assessed whether 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, 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.

These activities constituted four quarterly fire protection inspection samples as defined in IP 71111.05-05.

b. Findings

No findings were identified.

.2 Annual Fire Protection Drill Observation (71111.05A)

a. Inspection Scope

During RFO 1R23 on January 22, 2014, a callout of the onsite fire brigade occurred when a worker in the containment building observed smoke coming from machinery associated with the polar crane. The inspectors observed the fire brigade response to the situation. No flames were visible and power was quickly secured to the crane machinery. No lifts were in progress. Subsequent investigation by the licensee revealed that oil had dripped onto an electrical resistor and had heated up to cause the smoke. Functional checks of the crane were performed with no issues noted after the oil was cleaned up. The inspectors evaluated several attributes of fire brigade performance, and attended the critique held afterwards to ensure licensee staff identified deficiencies, openly discussed them in a self-critical manner, and took appropriate corrective actions. Specific attributes assessed during the response were:

- employment of appropriate firefighting techniques;
- sufficient firefighting equipment brought to the scene;
- effectiveness of fire brigade leader communications, command, and control; and
- utilization of pre-planned strategies.

Documents reviewed are listed in the Attachment.

These activities constituted one annual fire protection inspection sample as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors reviewed the licensee’s testing of the E-54 Component Cooling Water Heat Exchanger 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 compared the licensee’s observations to 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 design conditions and test conditions. Documents reviewed are listed in the Attachment.

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

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08P)

From January 21, 2014, through February 7, 2014, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring for any degradation of the primary coolant system (PCS), steam generator 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 inservice inspection sample as defined in IP 71111.08.

.1 Piping Systems ISI

a. Inspection Scope

The inspectors either observed or reviewed the following non-destructive examinations (NDEs) mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects were detected, to determine whether these were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- Ultrasonic Examination (UT) of a 4" Feedwater System Pipe-to-Elbow Weld (FWS-4-AWS-1S1-250) ;
- Dye Penetrant (PT) Examination of PCS, Pipe-to-Tee Weld, PCS-2-DRL-1H1-3; January 22, 2014;
- PT of PCS Tee-to-Reducer Weld, PCS-2-DRL-1H1-4; January 22, 2014;
- PT of PCS Tee-to-Pipe Weld, PCS-2-LDL-2B1-6; January 22, 2014;
- PT of PCS Pipe-to-Elbow, PCS-2-LDL-2B1-7; January 22, 2014;
- PT of PCS Elbow-to-Pipe, PCS-2-LDL-2B1-8; January 22, 2014;
- PT of PCS Pipe-to-Elbow, PCS-2-LDL-2B1-9; January 22, 2014;
- PT of PCS Elbow-to-Pipe, PCS-2-LDL-2B1-10; January 22, 2014;
- PT of PCS Pipe-to-Elbow, PCS-2-LDL-2B1-10A; January 22, 2014;
- PT of PCS Pipe-to-Elbow, PCS-2-LDL-2B1-3; January 22, 2014;
- PT of PCS Elbow-to-Pipe, PCS-2-LDL-2B1-4; January 22, 2014;
- PT of PCS Elbow-to-Pipe, PCS-2-LDL-2B1-10B; January 22, 2014;
- PT of PCS Pipe-to-Fitting, PCS-2-LDL-2B1-10C; January 22, 2014;
- PT of PCS Fitting-to-Pipe, PCS-2-LDL-2B1-10D; January 22, 2014;
- Visual Examination (VT-3) of Chemical and Volume Control (CVC) System, Pipe Restraint, CVC-2-LDL-2B2-21PR(H-1.7); and
- VT-3 of Engineered Safeguard System (ESS), Pipe Restraint, ESS-12-SIS-1LP-233PR (H713).

The inspectors reviewed the following examinations completed during the previous outage with relevant/recordable conditions/indications accepted for continued service to determine whether the acceptance was in accordance with ASME Code Section XI or an NRC-approved alternative.

- Indication (PT) Disposition of PCS "B" loop cold leg drain nozzle-to-pipe weld (PCS-2-DRL-1B1-1);

- Indication (UT) Disposition of weld PCS-2-LDL-2B1-1 (Weld 276), in the 2” cold leg letdown/drain line on PCS Loop 2B; and
- Indication (UT) Disposition of weld PCS-4-PRS-1P1-1 (Weld 165), in the Power-Operated Relief Valve (PORV) nozzle-to-pipe weld on the pressurizer.

The inspectors reviewed the following pressure boundary welds completed for risk-significant systems since the beginning of the last refueling outage to determine if the licensee applied the pre-service non-destructive examinations 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 the Construction Code and ASME Code Section IX.

- Weld repair/replacement of Class 2, Main Steam Safety Valve (Valve RV-0719);
- Weld repair/replacement of Class 1, CVC Check Valve (Valve CK-CVC2116); and
- Weld repair/replacement of Class 1, PCS Pipe-to-Valve Welds (Valve PRV-1072).

b. Findings

No findings were identified.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

A bare metal visual examination and a non-visual examination of the reactor vessel head was required this outage pursuant to 10 CFR 50.55a(g)(6)(ii)(D).

The inspectors observed the bare metal visual examination conducted on the reactor vessel head at each of the penetration nozzles 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, the inspectors determined:

- If the required visual examination scope/coverage was achieved and limitations (if applicable) were recorded in accordance with 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, whether the licensee entered the condition into their CAP 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, the inspectors determined:

- If the required examination scope (volumetric and surface coverage) was achieved and limitations (if applicable) were recorded in accordance with licensee procedures;
- If the UT examination equipment and procedures used were demonstrated by blind demonstration testing;

- For indications or defects that were identified, whether the licensee documented the conditions in examination reports and/or entered this condition into their CAP and implemented appropriate corrective actions; and
- For indications accepted for continued service, whether 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 IP attribute.

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

The inspectors performed an independent walkdown of the PCS and related lines in the containment, including the under vessel penetrations, which had received a recent licensee boric acid walkdown, and determined whether the licensee's Boric Acid Corrosion Control (BACC) visual examinations emphasized locations where boric acid leaks could 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.

- 13-PAL-0018; CV-1059, Pressurizer Spray Valve from Loop 2A has an Excessive Packing Leak;
- 12-PAL-0059; F-9, Boric Acid Filter has Boric Acid Buildup Coming from Under Insulation;
- 12-PAL-0086; Boric Acid Discovered on MO-3081 HPSI to Cold Leg-Hot Leg INJ [Injection] Mode Select Packing Area;
- 13-PAL-003; P-66B High Pressure, Safety-Injection Pump Boric Acid Evaluation; and
- 13-PAL-016; P-67B High Pressure, Safety-Injection Pump Boric Acid Evaluation.

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, "Corrective Action."

- CR-PLP-2012-02284; Boric Acid on MO-3062, HPSI TRN [Train] 2 Loop 2B;
- CR-PLP-2012-02450; Boric Acid on Top of ICI [In-Core Instrument] Flange Number 1; and
- CR-PLP-2012-05825; Boric Acid Deposits from Control Rod Drive (CRD) 24 on Head.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The inspectors observed the acquisition of eddy current testing (ET) data, interviewed ET data analysts, and reviewed documentation related to the steam generator (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, Steam Generator 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 TS, and 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 Eddy Current 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 evaluation (if applicable) and implemented 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 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment. The licensee generated CR-PLP-2014-01742, Code Case 770-1 Issue, in response to a concern identified by the inspectors and NRC staff in the Office of Nuclear Reactor Regulation (NRR) regarding examination of certain PCS penetration drain line welds. The inspectors also reviewed the licensee's response to this issue.

b. Findings

Failure to Complete Volumetric Examinations for Dissimilar Metal (DM) Butt Welds in Branch Connections

Introduction: A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50.55a(g)(6)(ii)(F)(3) was identified by the inspectors when licensee personnel failed to complete baseline volumetric UT examinations for nine dissimilar metal (DM) butt welds in the PCS that were fabricated from Inconel Alloy 82/182 weld metal and therefore were susceptible to PWSCC.

Description: During RFO 1R23, the inspectors and NRC staff from NRR identified that nine PCS penetration drain line welds had not been volumetrically examined with UT to complete the baseline examinations required by NRC regulations. These PCS welds were fabricated from Inconel Alloy 82/182 weld metal and were susceptible to PWSCC, which initiates from the inside of the weld surface. Operating experience had identified that Alloy 600/82/182 materials in DM welds exposed to primary coolant water or steam at normal operating conditions at PWR plants had cracked due to PWSCC. The NRC had issued several Bulletins and an Order since 2001 related to the occurrence of PWSCC in PCS components and welds containing Alloy 600/82/182. Absent volumetric baseline examinations, the inspectors were concerned that PWSCC may go undetected and lead to leakage or failure of these welds resulting in a LOCA.

For these nine DM butt welded branch connections (typically 2-inch nominal pipe diameter branch drain lines) in the Palisades PCS, eight were exposed to PCS cold leg operating temperatures and one was exposed to hot leg operating temperatures. The specific welds identified were as follows: PCS-30-RCL-1 A-11/2, PCS-30-RCL-1 A-5/2, PCS-30-RCL-1B-10/3, PCS-30-RCL-1B-5/2, PCS-30-RCL-2 A-11/2, PCS-30-RCL-2 A-11/3, PCS-30-RCL-2 A-5/2, PCS-30-RCL-2 B-5/2, and PCS-42-RCL-1HA-3/2. For these DM butt welds, both volumetric and visual examinations were required by ASME

CC N-770-1 based on the inspection category of A-2 (unmitigated butt welds exposed to hot leg temperatures) or the inspection category of B (unmitigated butt welds exposed to cold leg temperatures). Title 10 CFR 50.55a(g)(6)(ii)(F)(3) required that baseline examinations for welds in CC N-770-1, Table 1, Inspection Items A-1, A-2, and B, be completed by the end of the next refueling outage after January 20, 2012. Palisades completed a refueling outage in May of 2012, without completing the required volumetric examination of these nine welds.

Additional clarification was provided in 10 CFR 50.55a(g)(6)(ii)(F)(2), which stated, in part, "...All other butt welds that rely on Alloy 82/182 for structural integrity shall be categorized as Inspection Items A-1, A-2, or B until the NRC Staff has reviewed the mitigation and authorized an alternative CC Inspection Item for the mitigated weld..." The licensee incorrectly believed that this requirement only applied to Alloy 82/182 welds that had undergone some type of mitigation activity. Additionally, the licensee had incorrectly concluded that the absence of a figure for, or any reference to, branch connection welds in CC N-770-1 demonstrated that the applicability of CC N-770-1 was limited to circumferential butt welds. Specifically, the licensee stated that the term "butt weld" referred to circumferential butt welds in piping systems, not branch connection welds. The inspectors reviewed the original construction code and determined that these welds were fabricated as butt welded branch connections, and as such were subject to the augmented inspections required by 10 CFR 50.55a(g)(6)(ii)(F)(2).

In response to the NRC staff concern with the lack of a volumetric examination to confirm the absence of PWSCC in these welds, the licensee entered this issue into their CAP as CR-PLP-2014-01742, NRC Question on Whether Hot and Cold Leg Branch Connection Welds are In Scope of ASME Code Case N-770-1, dated February 27, 2014. On February 25, 2014, the licensee submitted a request for relief (ML14056A533) to the NRC to allow substitution of a visual and dye penetrant surface examination of these welds as an alternative to volumetric examinations. As part of this request, the licensee submitted the results of a flaw growth analysis for the hot leg drain nozzle butt weld and concluded that the ASME Code flaw acceptance criteria would be met for 60 effective full power years for circumferential cracks and 34 effective full power years for axial cracks. As of January 2014, the Palisades plant had operated for approximately 26.2 effective full power years. After various requests for additional information and discussions between the licensee and the NRC, the NRC granted verbal relief on March 13, 2014 (ML14073A274). This verbal relief stated the licensee could implement the proposed alternative to 10 CFR 50.55a(g)(6)(ii)(F), which included a commitment to perform enhanced leakage monitoring during the current operating cycle, and perform the required volumetric examinations during the next RFO.

Analysis: The inspectors determined that the licensee's failure to complete volumetric examinations on the nine DM butt welded PCS branch connections fabricated with Alloy 82/182 weld metal as required by ASME CC N-770-1 was a performance deficiency that warranted a significance evaluation.

The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the finding was associated with the Equipment Performance (Reliability) attribute of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors also determined that if left uncorrected the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, absent NRC identification, the failure to complete volumetric examinations on the nine DM butt welded PCS branch connections fabricated with Alloy 82/182 weld metal could have allowed PWSCC to remain in service that could propagate and result in a LOCA. The inspectors performed a Phase I SDP screening using IMC 0609, Appendix A, "The Significance Determination Process (SDP) for Findings At-Power," Exhibit 1, "Initiating Events Screening Questions." The inspectors answered the Phase I SDP "LOCA Initiators" Questions A1 and A2 'No' because undetected cracks, if present, were not yet through-wall and did not challenge the structural integrity of the welds. Therefore, this finding screened as having very low safety significance (Green).

The finding had an associated cross-cutting aspect in the Evaluation (P.2) component of the Problem Identification and Resolution cross-cutting area because the licensee failed to thoroughly evaluate this issue to ensure that the resolution addressed causes and extent of condition commensurate with safety. Specifically, when determining the applicability of CC N-770-1, the licensee failed to thoroughly evaluate the scope of welds susceptible to PWSCC that required volumetric examination commensurate with the safety significance of this issue.

Enforcement: Title 10 CFR Part 50.55a(g)(6)(ii)(F), "Examination requirements for class 1 piping and nozzle dissimilar-metal butt welds," requires, in part, that "(1) Licensees of existing, operating pressurized-water reactors as of July 21, 2011, shall implement the requirements of ASME CC N-770-1, subject to the conditions specified in Paragraphs (g)(6)(ii)(F)(2) through (g)(6)(ii)(F)(10) of this section, by the first refueling outage after August 22, 2011." Title 10 CFR Part 50.55a(g)(6)(ii)(F)(3) requires, in part, that "Baseline examinations for welds in Table 1, Inspection Items A-1, A-2, and B, shall be completed by the end of the next refueling outage after January 20, 2012." ASME CC N-770-1, "Examination Categories," requires, in part, that "volumetric examinations" be performed for "Parts" (e.g., welds) defined as "Inspection Items" "A-1," "A-2" and "B." Inspection Item A-2 was defined as "Unmitigated butt weld at Hot Leg operating temperature (-2410) ≤ 625°F (329°C)," and Inspection Item B was defined as "Unmitigated butt weld at Cold Leg operating temperature (-2410) ≥ 525°F (274°C) and < 580°F (304°C)."

Contrary to the above, the licensee completed a refueling outage in May 2012 (first refueling outage after August 22, 2011, and the next refueling outage scheduled after January 20, 2012) without performing required baseline volumetric examinations for nine PCS DM butt welded branch connections. One of these nine welds was an unmitigated DM butt weld that was exposed to hot leg operating temperatures, which would be classified as a Category A-2 item. The remaining eight welds were unmitigated DM butt welds exposed to cold leg operating temperatures, which would be classified as a Category B item.

As part of the licensee's corrective actions, a relief request was requested, which included a commitment to perform enhanced leakage monitoring during the current operating cycle and perform the required volumetric examinations during the next refueling outage.

Because this violation was of very low safety significance and because the issue was entered into the licensee's CAP as CR-PLP-2014-01742, NRC Question on Whether Hot and Cold Leg Branch Connection Welds are In Scope of ASME Code Case N-770-1, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000255/2014002-02, Failure to Complete Volumetric Examinations for DM Butt Welds in Branch Connections)**

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On March 5, 2014, the inspectors observed a crew of licensed operators in the plant's simulator during just-in-time training for diluting to criticality for re-start from RFO 1R23. The inspectors determined whether operator performance was adequate, evaluators were identifying and documenting crew performance problems, and 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;
- the ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- control board manipulations;
- crew pre-job briefing; and
- oversight and direction from supervisors.

The performance in these areas was compared to pre-established operator action expectations, procedure compliance, and successful critical task completion requirements. Documents reviewed are listed in the Attachment.

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11.

b. Findings

No findings were identified.

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

a. Inspection Scope

On January 23, 2014, the inspectors observed operations staff conducting heightened risk activities in the control room during RFO 1R23. Specifically, the operators were draining the PCS to a reduced inventory condition for installation of steam generator nozzle dams. This was an infrequently performed task or evolution that required heightened awareness across the site and coordination amongst operators at various stations outside the control room. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;

- the ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board and control rod manipulations; and
- oversight and direction from supervisors.

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

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

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:

- CVC System; and
- Auxiliary Feedwater Floor Drains.

The inspectors reviewed events including those in which 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.

This inspection constituted two 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:

- Maintenance on electrical bus 1-C and inside electrical panel D-11A;
- Heavy load lifts inside turbine building and containment during RFO 1R23;
- SW system isolation issues associated with replacement of SW system piping and valves; and
- PCS cold leg plug installation for Alloy 600 work during RFO 1R23.

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 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 are listed in the Attachment.

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:

- Seismic Qualification of RR1 Relays;
- Operability of Spent Fuel Pool Region II Due to Criticality Calculation Questions;
- 1-1 Emergency Diesel Generator Failure to Start;
- Part 21 Issued for 480 Volt ABB Breakers; and
- Evaluation of Foreign Material Left in Reactor Vessel Following RFO 1R23.

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 the 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.

This operability inspection constituted five samples as defined in IP 71111.15-05.

b. Findings

(Unresolved Item) Spent Fuel Pool Region II Criticality Analysis

Introduction: The inspectors identified an Unresolved Item (URI) regarding assumptions used in the criticality analysis for Region II of the licensee's spent fuel pool. Specifically, several assumptions in the applicable criticality analysis, which supported compliance with TSs and NRC regulations for criticality, did not appear to bound the characteristics of some fuel assemblies stored in Region II of the spent fuel pool.

Description: On November 5, 2013, the licensee initiated CR-PLP-2013-04775, Issues Identified with Region II of Spent Fuel Pool Critically Analysis, which documented that the spent fuel pool criticality analysis was not updated following a power uprate that had been implemented in 2004. This was identified during the licensee's review of industry operating experience documenting a similar issue at a different power plant. The licensee identified the following concerns with the criticality analysis for Region II of the spent fuel pool: 1) the assumed fuel temperature depletion parameter did not appear to bound the actual temperature for Batch A fuel, 2) the assumed cycle boron concentration did not appear to bound the actual cycle boron concentration after Cycle 20, and 3) the assumption that all Promethium-149 has decayed to Samarium-149 prior to placement of fuel into the spent fuel pool did not appear to be directly translated into site procedures.

These concerns ultimately focused on whether fuel had achieved adequate burnup prior to placement in Region II of the spent fuel pool. The criticality analysis stated Batches A, B, and C fuel from Cycle 1 would not qualify for storage in Region II of the spent fuel pool due to extremely low burnup. However, Batch A fuel had been stored in Region II since a spent fuel pool re-rack project in 1987. Most of the Batch A fuel was relocated to dry storage in 1994 and 1995, but nine Batch A fuel assemblies currently remain stored in Region II. As a result of the assumptions that appeared to not bound actual conditions, Operations requested an Operability Evaluation to further evaluate the issue.

Operability Evaluation CR-PLP-2013-04775 was assigned on November 5, 2013, and completed on December 5, 2013. The inspectors reviewed the Operability Evaluation along with staff from the Spent Fuel Team in the Office of Nuclear Reactor Regulation

(NRR), Division of Safety Systems (DSS). On March 20, 2014, the NRC discussed the following questions regarding the Operability Evaluation with the licensee:

- The licensee compared the post-uprate hot leg temperature to the reactor core temperature in the analysis of record to justify that the analysis was bounding. However, the core temperature was hotter than the hot leg temperature, thus the Operability Evaluation did not appear to demonstrate that the existing core temperature was bounded by the core temperature in the analysis of record.

- Technical Specification Table 3.7.16-1 did not appear to ensure compliance with 10 CFR 50.68, which addressed spent fuel criticality, or TSs 4.3.1.3.a or 4.3.1.3.b, both of which addressed design assumptions in the Region II fuel storage racks.

- The methodology used in the development of the analysis of record contained non-conservatisms that appear to be mitigated by design margins that were already credited elsewhere in the Operability Evaluation.

On March 25, the NRC questions were entered into the licensee's CAP as assignments 6 and 7 of CR-PLP-2013-04775 with due dates of April 8. At the conclusion of the inspection period, the NRC staff was waiting to review the responses to the questions provided on March 20. This is an URI pending NRC review of the requested additional information. **(URI 05000255/2014002-03, Spent Fuel Pool Region II Criticality Analysis)**

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- Station Battery ED-01 Cell #1 Replacement;
- Control Room Heating, Ventilation and Air Conditioning (HVAC) Chiller, VC-10, Relay Replacement After Tripping;
- CV-2155, Makeup Stop Valve, Suspected Leakby; PCS Unidentified Leakage Calculations;
- P-18A, Diesel Fuel Oil Transfer Pump and T-10A, Diesel Fuel Oil Tank Repairs and Inspections During RFO 1R23; and
- P-50C, 'C' Primary Coolant Pump, Impeller Replacement During Refueling Outage.

These activities were selected based upon the SSC'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 (e.g., 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.

This inspection constituted five 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 a scheduled refueling outage that began on January 19, 2014, and continued through March 18, 2014. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, startup and heatup activities, and identification and resolution of problems associated with the outage.

One of the planned refueling outage activities of particular NRC interest was a foreign object search and retrieval (FOSAR) activity in the reactor vessel. While licensees routinely inspect for foreign material in plant systems and implement controls to prevent the introduction of debris into plant systems, the licensee has in the past identified broken pieces of primary coolant pump (PCP) impellers in the reactor vessel. As a result of a PCP-C vibration transient on October 29, 2011, the licensee suspected a piece of impeller might have broken off and entered the reactor vessel.

Issues with PCP impellers at Palisades date back to 1971 when the impeller for PCP-A was weld-repaired and reinstalled due to damage from foreign material. Below is a timeline of continued issues with PCP impellers.

1983: The licensee identified and removed a piece of broken impeller from under the reactor core barrel during core-offload as part of refueling outage activities. The licensee inspected all of the PCPs and noted the piece originated from PCP-C. The damaged PCP-C impeller was replaced with a new impeller in early 1984.

1984: The newly installed PCP-C impeller failed due to improper assembly and required replacement. The licensee acquired an impeller from another plant, trimmed the impeller diameter to the proper size, and installed the new impeller.

1999: The site commenced a project to refurbish or replace the four PCPs.

The PCP-A impeller was removed for replacement with a spare unused impeller. The removed impeller had cracking on two of the five vanes that was attributed to inadequate post-weld heat treatment in 1971, and the impeller was weld-repaired for future use.

2001: The weld-repaired impeller from PCP-A was installed in PCP-B. This was the first impeller replacement for PCP-B and the removed impeller had cracking in three of the five vanes. The removed impeller was weld-repaired for future use.

2003: The licensee removed the trimmed impeller from PCP-C for replacement and noted extensive damage such that repair was not viable. The impeller was replaced with the refurbished impeller that had been removed from PCP-B.

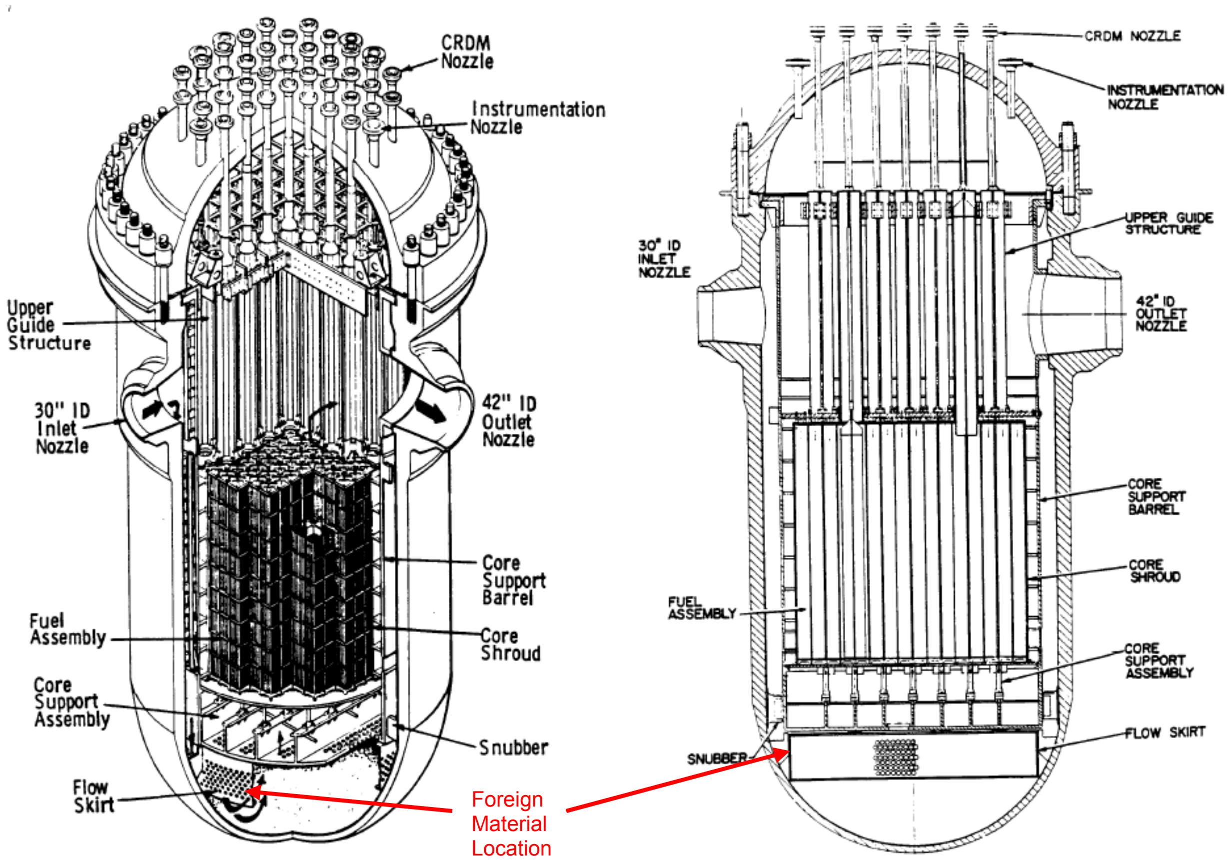
2007: The licensee identified and removed two impeller pieces from the reactor vessel.

2009: The original impeller in PCP-D was removed and replaced with a newly manufactured impeller. The original impeller was subsequently inspected and found to have recirculation damage, but no cracking.

2014: The licensee removed the impeller from PCP-C and replaced it with a newly manufactured impeller. The removed impeller had missing portions in two impeller vanes.

The 2014 refueling outage included the removal of the core barrel to support more comprehensive reactor vessel inspections than can typically be conducted. This activity was pre-planned for reasons not related to the pump impeller concern, but coincidentally allowed for a more thorough inspection for foreign material in the reactor vessel. The licensee anticipated finding the suspected broken piece from the PCP-C vibration event in October 2011 in the vessel. The reactor vessel FOSAR activities identified two pieces of broken impeller; one piece was removed from the vessel and the other piece was lodged between the reactor vessel and the bottom corner of the flow skirt. The licensee attempted to remove the lodged piece using several methodologies, including pulling using vice grips and pushing using hydraulic tools. Despite the application of approximately 3,000 pounds per square inch (psi) of force, the piece did not move.

The reactor vessel is shown in the figures below. Four PCPs circulate water through the PCS. After the water has passed through the steam generators and transferred heat to the secondary system, the water is pumped by the PCPs through the PCS cold legs and into the reactor vessel. In the included figures that depict the Palisades PCS, one cold leg (inlet nozzle) and one hot leg (outlet nozzle) are shown for the purpose of simplicity. In actuality, there are four cold legs and two hot legs. Water enters the reactor vessel via the cold legs and flows down between the reactor vessel wall and the core support barrel. Near the bottom of the vessel is a flow skirt that contains many small holes that most of the water flows through. Some water also passes below the flow skirt into the bottom of the vessel. After flowing through or under the flow skirt, the water then flows up into the active fuel region to remove heat from the nuclear fuel. After flowing through the fuel region, the water exits the vessel through the hot legs and into the steam generators.



View of the impeller piece looking down from above the flow skirt

View of the impeller piece looking up from below the flow skirt.

Since the licensee could not remove the impeller piece from the vessel, Operability Evaluation CR-PLP-2014-01510 was developed to evaluate the operability of the reactor vessel. The piece was tapered in thickness from $^{3}/_{16}$ inches to roughly one inch wide, and the gap between the vessel wall and flow skirt where the piece was wedged was up to ½ inch wide. The piece was not blocking any of the flow holes through the flow skirt. Plant history has shown that prior broken impeller pieces that passed through the gap were found at the bottom of the vessel. The licensee performed a fluid dynamics analysis to determine the forces that would act on the piece during plant operation, which concluded that the maximum force would be a 350 pound lift force. The site then performed a structural analysis to determine the effects of the piece and hydraulic forces on the reactor vessel and flow skirt. Heatup and cooldown effects were considered and the flow skirt and vessel were determined to move together such that the gap size would remain constant. A fracture analysis was performed to determine if the piece would break up into smaller pieces during the operating cycle. The analysis assumed several initiating crack sizes in the piece, all of which determined that the crack growth rate would reduce and essentially stop once the crack depth approached 75 percent of the thickness of the piece. Based on the results of the analyses, the licensee concluded that the piece would not move, would not break up, would not impede PCS flow, and would not affect the pressure-retaining capability of the reactor vessel. The analyses were performed as a joint effort between the licensee and equipment vendors. The licensee's Operability Evaluation concluded the reactor vessel was operable with the impeller piece wedged between the reactor vessel and the flow skirt.

The licensee concluded that the cause of the repeated impeller failures was fatigue-related effects from the operation of the PCPs in conditions beyond the maximum flow rate and below the minimum net positive suction head recommendations as described in design documentation. These conditions are present when operating only one or two PCPs (one on each loop) during reduced temperatures and pressures (i.e., during startup and shutdown activities). Cyclic pressure pulses and stresses are created under these reduced pressure conditions that act on the leading edges of the impellers, which can ultimately lead to impeller vane cracking and the break-off of small impeller pieces. The licensee determined, based on metallurgical examination of a previous impeller piece that broke off and the mechanism by which the cracks propagated, that weld refurbished impellers were particularly susceptible to degradation. At normal operating temperature and pressure, there is adequate net positive suction head on all PCPs, so these additional stresses are not present.

The inspectors and NRC staff from headquarters conducted an in-depth independent review of the analyses forming the basis for the licensee's conclusions. The independent review included:

- The licensee's analytical basis for why the wedged impeller fragment was expected to remain in place;
- The licensee's determination that the impact of the impeller fragment wedged between the reactor vessel and the flow skirt did not exceed the structural integrity of the vessel wall or the flow skirt support welds;
- The licensee's analysis for why the wedged impeller fragment was not expected to break into smaller pieces and in the unlikely scenario that it did, the impact of the pieces on fuel cooling, fuel cladding, and the reactor vessel structure;
- The licensee's assessment of the potential for corrosion at the interface of the wedged impeller fragment, reactor vessel, and flow skirt; and
- The licensee's assessment of a worst case scenario accident that could result in the impeller piece impacting the reactor vessel or affecting fuel integrity.

Based on this independent review, the NRC concluded that the impeller piece did not pose a threat to safe operation of the reactor and reactor vessel. Because the PCP-C impeller was replaced with a new impeller this outage, PCP-B was the only pump that remained in service with a refurbished impeller that was more susceptible to the fatigue-related failures that have been observed. The licensee ensured that PCP-B was not one of the first two PCPs started following the Spring 2014 refueling outage, which did not expose PCP-B to the susceptible pressure and flow conditions. However, because PCP-B continues in service with potential impeller vane cracks there remains a potential for impeller pieces to break off. The inspectors and NRC staff recognized this concern and did not identify any immediate safety concerns, in part due to the extensive operating experience with broken impeller pieces. However, a review of the licensee's evaluation to justify continued operation of PCP-B with a potentially cracked impeller continues. Additionally, the inspectors continue to review the licensee's corrective actions to date and going forward to determine whether the licensee plans to eliminate the known susceptibility of impeller pieces breaking off. The inspectors planned to document these ongoing reviews in Section 4OA2 of future inspection reports, in accordance with IP 71152.

Documents reviewed are listed in the Attachment.

This inspection constituted one refueling outage sample as defined in IP 71111.20-05.

b. Findings

(1) Introduction of Foreign Material into the SW System

Introduction. A finding of very low safety significance (Green) and an associated NCV of TS 5.4.1, "Procedures," was self-revealed when foreign material that consisted of an inflatable bladder was introduced on two separate occasions into the SW system return header. These events occurred during maintenance on the SW system during RFO 1R23.

Description. During RFO 1R23, work was being performed on the SW system to address some leaks that had developed during the operating cycle. Two of the leaks were located downstream of the 'A' Component Cooling water (CCW) heat exchanger; one on a pipe elbow of the 16" outlet piping and the other on a manual valve in the 4" outlet line. Repairs consisted of a replacement of the 16" elbow and the 4" manual valve. The supply of SW was isolated to perform the work. No isolations were available on the return header of the work areas. In addition, the 'B' CCW heat exchanger, located below 'A', was required to remain in service during the work. As a result, while attempting to perform maintenance, workers noted excessive splashing of water from the outlet of the 'B' CCW heat exchanger up into the work areas. Attempts to resolve the issue by throttling flow were not effective. The decision was made to place an inflatable rubber bladder into the 16" pipe down far enough past both work areas to block the water. Steps were added to the WO work instructions to document when the bladder was installed and removed, a lanyard was attached to the bladder, and attempts were made to seat and inflate the bladder. During the bladder installation, Operations repositioned the SW return valve from containment. This action added to the suction effects already present in the piping near the work area that were caused by the piping arrangement and flow from the 'B' CCW heat exchanger. As a result, the bladder ripped from the lanyard and was sucked into the return header. The bladder was later found outside the system in the discharge basin. No impact to plant systems was noted. Later in the outage, with decay heat load reduced, another attempt was made to throttle flow to facilitate work. Day shift Operations staff concluded that use of a bladder would likely not be necessary. However, during field work at night, maintenance personnel decided a bladder should be used. While they had informed night shift Operations of the possibility of needing one before work commenced, night Operations staff were unaware that day shift had concluded bladder use should be avoided (based on the past experience) and that flow adjustments alone would likely be successful. As a result, maintenance workers attempted to install another bladder and it too was sucked into the system return header. The bladder was not found and is believed to be either in the return piping or Lake Michigan. At the end of the inspection, no impact to plant systems had been noted, and a comprehensive system test was successfully completed to demonstrate SW system operability.

The inspectors observed some of the field activities, and reviewed the work instructions and procedure EN-MA-118, "Foreign Material Exclusion." Section 5.2[1] of EN-MA-118 stated, in part, that planners and procedure writers should evaluate FME considerations for work activities and include job-specific FME controls in work instructions and procedures. Section 5.2[6] stated, in part, that during the planning stage, the planner should designate the FME Zone type, risk level, pathways to FME sensitive equipment (based on Piping & Instrumentation Diagram reviews), and work practice restrictions as applicable in all work packages. The inspectors determined that these steps were not followed as no controls were designated in the work instructions regarding how equipment manipulations (cycling of valves/flow in the system) could affect the bladder. Additionally, there was no formal assessment on the type of bladder being used and potential impacts on the SW system or FME Zone type (contractor personnel had noted that they typically did not use the bladder under vacuum; and insertion of a large bladder was beyond the scope of the initial FME evaluation that only considered cutting/welding work). The inspectors' review also revealed that the FME checklist in the work instructions was not updated when the decision was made to introduce a bladder into the system, which could have highlighted the need for further controls or re-evaluation.

The licensee entered this issue into their CAP as CR-PLP-2014-00715, Vacuum was So Great that Bladder was Ripped Off Lanyard and Lost in Piping, and CR-PLP-2014-01176, FME Bladder was Lost in Pipe Due to Excessive Vacuum. As part of the licensee's corrective actions, the work was completed using system flow adjustments alone.

<u>Analysis</u>. The failure to follow EN-MA-118, "Foreign Material Exclusion," during work on the SW system was a performance deficiency that warranted a significance evaluation.

The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because if left uncorrected, the finding would have the potential to lead to a more significant safety concern. The fact that there was a repeat occurrence of foreign material introduction into the SW system along with several other observations of inadequate FME control implementation, led the inspectors to conclude a programmatic deficiency existed.

Additionally, the inspectors determined that the finding was associated with Configuration Control attribute of the Initiating Events cornerstone and adversely impacted the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The significance of the finding was assessed utilizing IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," Attachment 1. Based on Checklist 3, "PWR [Pressurized Water Reactor] Cold Shutdown and Refueling Operation RCS [Reactor Coolant System] Open and Refueling Cavity Level < 23' or RCS Closed and No Inventory in Pressurizer Time to Boiling < 2 hours," following the loss of the first bladder, and Checklist 4, "PWR Refueling Operation: RCS level > 23' OR PWR Shutdown Operation with Time to Boil > 2 hours And Inventory in the Pressurizer," following the loss of the second bladder, the inspectors determined none of the mitigation capabilities were lost. Additionally, utilizing Table 1 of IMC 0609, Appendix G, the inspectors determined there was no loss of control. As a result, the finding screened as having very low safety significance (Green).

This finding had an associated cross-cutting aspect in the Work Management (H.5) component of the Human Performance cross-cutting area because the organization did not implement a process of planning, controlling, and executing work activities such that nuclear safety was the overriding priority. The work process did not include the identification and management of risk commensurate with the work and the need for coordination with different groups or job activities.

<u>Enforcement</u>: Technical Specification 5.4.1, "Procedures," requires, in part, implementation of the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Section 9 of Regulatory Guide 1.33 states, in part, that maintenance that can affect the performance of safety-related equipment should be properly preplanned and performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances.

Contrary to the above, on January 27, 2014, and February 8, 2014, the licensee failed to implement the requirements of procedure EN-MA-118, "Foreign Material Exclusion," during work on the SW system. As a result, an inflatable bladder twice entered the return header of the system, which had the potential to affect decay heat removal and spent fuel pool cooling during a refueling outage.

As part of their corrective actions, the licensee evaluated the condition and based on successful SW system testing and no impact noted on system performance, determined the system was operable.

Because this violation was of very low safety significance and because the issue was entered into the licensee's CAP as CR-PLP-2014-00715, Vacuum was So Great that Bladder was Ripped off Lanyard and Lost in Piping, and CR-PLP-2014-01176, FME Bladder was Lost in Pipe Due to Excessive Vacuum, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000255/2014002-04, Introduction of Foreign Material Into the SW System)**

(2) <u>Failure to Follow Procedures During Reactor Vessel Head Lift</u>

<u>Introduction</u>: A finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when licensee personnel failed to follow maintenance procedure RFL-R-16, "Reactor Vessel Closure Head Installation." Specifically, during the reactor vessel head lift on March 5, 2014, for reinstallation onto the vessel flange, workers failed to identify an interference with the reactor head lift structure, which caused the head to

impact a jack screw on the structure. The total load was unexpectedly and unknowingly increased to approximately 283,000 pounds, which was greater than the procedural maximum polar crane load rating of 270,000 pounds.

Description: On March 5, 2014, contract and site personnel were lifting the reactor vessel head off of the support pedestals for reinstallation onto the vessel flange using the containment polar crane. The work group had successfully completed the first section of procedure RFL-R-16, "Reactor Vessel Closure Head Installation," to lift the head 3 inches and hold for 5 minutes to verify a steady lift with a load cell reading of 254,500 pounds. After completing this 5-minute hold, the work group continued with the head lift evolution, but called an all-stop when workers identified that the head had made contact with the reactor head lift structure that surrounded the reactor head pedestals and caused a large increase in the load cell reading. The workers who identified the increased load cell reading of approximately 283,000 pounds did not realize that this exceeded the maximum load rating of the polar crane although this maximum load rating was specified in an RFL-R-16 procedure Caution Note. The head was then lowered slightly until the interference of the structure was removed and the load cell had a stable reading of approximately 255,000 pounds. It was then decided among the crew members inside containment and via headset to the project manager in an outside trailer to continue with the head lift.

Palisades Maintenance Procedure RFL-R-16, "Reactor Vessel Closure Head Installation," Section 5.10, provided the directions for moving the reactor vessel head from the support pedestals to the flange. Step 5.10.3.c.2 of RFL-R-16 directed the crew to observe for interference/obstruction from the lift structure while moving the head clear of the support pedestals. Procedure RFL-R-16 also provided steps to adjust the jack screw height as needed to prevent contact with the head. These steps were not completed appropriately. Section 5.10 contained a Caution Note that stated, "MAXIMUM polar crane load rating limited to 270,000 pounds," and a separate note that stated, "any lifting or lowering operation may be stopped immediately as required due to unexpected circumstances." The step prior to the 5 minute hold directed that the load cell be monitored and maintained less than or equal to 270,000 pounds as the head was raised from the support pedestals. If the load cell reading indicated greater than 270,000 pounds, RFL-R-16 directed workers to immediately stop the lift, lower the load until the load cell reads less than the maximum, and notify the design engineer, shift manager, and shift outage director for approval to proceed.

The reactor vessel head was subsequently placed on the vessel flange. Following completion of this activity, and after workers realized that the maximum load limit of the polar crane had been exceeded, the crane was inspected by the vendor and site personnel and found to be in a safe operating condition. It was determined from the load rating design calculation of the polar crane that the actual design rating was 300,000 pounds.

The licensee entered this issue into their CAP as CR-PLP-2014-01903, Reactor Head Flange Contacted Jacking Screw While Raising It Off the Head Stand. A Level 1 Human Performance Evaluation was completed, which identified the aforementioned apparent and contributing causes of the event. Immediate actions taken as a result of this event included crew stand downs on crane and rigging practices and walkdowns of all remaining lifts to verify no interferences or obstructions were present. Longer term corrective actions included the review of the reactor head reinstallation procedure to determine whether changes could be incorporated to prevent recurrence and consideration of including sign-offs for supervisor level walkdowns of lifts prior to them commencing.

Analysis: The inspectors determined that the failure to follow procedure RFL-R-16 was a performance deficiency that warranted a significance determination.

The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, “Issue Screening,” because the finding was associated with the Human Performance attribute of the Barrier Integrity cornerstone and adversely impacted the cornerstone objective of providing reasonable assurance that physical design barriers (fuel cladding, reactor coolant system, and containment) protect the public from radionuclide releases caused by accidents or events. Additionally, the inspectors determined that the performance deficiency could reasonably be viewed as a precursor to a significant event and that if left uncorrected the performance deficiency could have the potential to lead to a more significant safety concern. Specifically, the operability of the containment polar crane was required to be evaluated and the reactor vessel head was required to be inspected after the event occurred to verify no significant damage was caused, and the evolution as conducted would not have precluded operation of the polar crane above its actual load limit.

The finding was screened in accordance with IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” Attachment 1, “Shutdown Operations Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs.” The finding screened as having very low safety significance (Green) based on not requiring a quantitative assessment after reviewing the five shutdown safety functional areas in Checklist 3, “PWR Cold Shutdown and Refueling Operation RCS Open and Refueling Cavity Level < 23' Or RCS Closed and No Inventory in Pressurizer Time to Boiling <2 hours.”

This finding had an associated cross-cutting aspect in the Challenge the Unknown [H.11] component of the Human Performance cross-cutting area. Specifically, human performance investigations identified that the workers exhibited a lack of rigor when performing no interference verifications prior to and during the reactor head lift, and an inadequate “stop when unsure” mentality when assessing the situation before continuing with the head lift. In addition, the workers and supervisors for this task did not understand that the load cell increase exceeded the procedural maximum value and did not inform decision-makers outside of the immediate work area to validate it was safe to proceed with the evolution.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality shall be accomplished in accordance with instructions, procedures, and drawings of a type appropriate to the circumstances. Palisades Maintenance Procedure RFL-R-16, “Reactor Vessel Closure Head Installation,” Section 5.10, provided the directions for moving the reactor vessel head from the support pedestals to the flange. Step 5.10.3.c.2 of RFL-R-16 directed the crew to observe for interference/obstruction from the lift structure while moving the head clear of the support pedestals. Procedure RFL-R-16 also provided steps to adjust the jack screw height as needed to prevent contact with the head. Section 5.10 contained a Caution Note that stated, “MAXIMUM polar crane load rating limited to 270,000 pounds,” and a separate note that stated, “any lifting or lowering operation may be stopped immediately as required due to unexpected circumstances.”

Contrary to the above, workers failed to follow procedure RFL-R-16, “Reactor Vessel Closure Head Installation,” during the reactor vessel head lift on March 5, 2014. The workers failed to identify an interference with the reactor head lift structure, which caused the head to impact a jack screw on the structure, and increased the total load to approximately 283,000 pounds, which was greater than the procedural maximum polar crane load rating of 270,000 pounds.

Immediate actions taken as a result of this event included crew stand downs on crane and rigging practices and walkdowns of all remaining lifts to verify no interferences or obstructions were present. Long-term corrective actions included the review of the reactor head reinstallation procedure to determine whether changes could be incorporated to prevent recurrence and consideration of including sign-offs for supervisor level walkdowns of lifts prior to them commencing.

Because this violation was of very low safety significance and because this issue was entered into the licensee’s CAP as CR-PLP-2014-01903, Reactor Head Flange Contacted Jacking Screw While Raising It Off the Head Stand, this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000255/2014002-05, Failure to Follow Procedures During Reactor Vessel Head Lift)**

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:

- T-302, 1-1 Emergency Diesel Generator Overspeed Trip Test (Routine);
- QO-15, ‘A’ Component Cooling Water Pump Quarterly Surveillance Test (IST);
- DWO-1, Operator’s Daily/Weekly Items Modes 1, 2, 3, and 4 (Routine);
- WO 52435724 and WO 52435755, LT-0105 and LT-0106 Calibrations (Routine);
- RO-32-69, Local Leak Rate Test Procedure for Penetration MZ-69 (Containment Isolation Valve);
- RO-141, Containment Sump Check Valves Inservice Test (IST);
- RO-98, LPSI and Containment Spray Comprehensive Pump Test and Check Valves Test (IST);
- RO-144, Comprehensive Pump Test Procedure for SW Pumps P-7A, P-7B, P-7C (IST);
- RO-65, HPSI, Trains 1 and 2, and Hot Leg Injection Check Valve Test and Cold Leg/Hot Leg Flow Balance Test (IST); and
- RT-8C, Engineered Safeguards System – Left Channel (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 IST 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 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.

This inspection constituted four routine surveillance testing samples, five IST samples and one containment isolation valve inspection sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Occupational Radiation Safety and Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted a partial sample as defined in IP 71124.01-05.

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

- 1R23 Refueling Outage Insulation Activities; and
- Refuel Project: Incore Instrumentation (ICI) Removal/Installation.

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 whether the licensee had established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that could 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.

.2 Instructions to Workers (02.03)

a. Inspection Scope

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:

- 1R23 Refueling Outage Insulation Activities;
- Refuel Project: ICI Removal/Installation.

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 setpoints were in conformance with survey indications and plant policy.

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.

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

- RWP 20140421; 1R23 Insulation Activities;
- RWP 20140429; Refuel Project: ICI Removal/Installation.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including the 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.

b. Findings

No findings were identified.

.4 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.

b. Findings

No findings were identified.

.5 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to radiation protection work 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.

b. Findings

No findings were identified.

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

This inspection constituted a partial sample as defined in IP 71124.02-05.

.1 Verification of Dose Estimates and Exposure Tracking Systems (02.03)

a. Inspection Scope

The inspectors reviewed the assumptions and basis (including dose rate and person-hour estimates) for the current annual collective exposure estimate for reasonable accuracy for select ALARA work packages. The inspectors reviewed applicable procedures to determine the methodology for estimating exposures from specific work activities and the intended dose outcome.

b. Findings

Failure to Maintain Radiation Exposure ALARA During Control Rod Drive Mechanism (CRDM) 24 Repairs

Introduction: A finding of very low safety significance (Green) was self-revealed due to unplanned and unintended occupational collective radiation dose that was received as a result of deficiencies in the licensee’s Radiological Work Planning and Work Execution Program. Specifically, the licensee failed to properly incorporate ALARA strategies and insights while planning and executing work activities on CRDM 24 during an August 2012 maintenance outage. This issue was originally identified as Unresolved Item (URI) 05000255/2013005-04, “Evaluation of Dose Received by Workers Repairing CRDM 24.”

Description: During an August 2012 maintenance outage, numerous work tasks were performed, including repairs to the CRDM 24 housing. The initial dose estimate for this RWP was 2.950 Rem. The actual dose incurred was 26.563 Rem. The licensee provided data that was incomplete in several areas. However, the inspectors concluded that a nominal 8.5 person-Rem of exposure was beyond the licensee's ability to foresee and correct and was attributable to emergent work. Specifically, the dose attributed to the necessity to inspect additional CRDM housings as part of the licensee's extent of condition review was discounted from regulatory consideration by the inspectors. The inspectors also excepted from regulatory consideration the dose attributable to implementation of ALARA dose reduction strategies, such as the installation of additional shielding in the work area. However, the inspectors concluded that several work planning and work execution issues were within the licensee's ability to foresee and correct, and therefore, should have been prevented. Specific examples included ultrasonic testing exams that were re-performed due to insufficient or inadequate initial exams, poor coordination of shielding installation and removal that necessitated field re-work, and inadequate mock-up testing that resulted in in-field work activities that contributed to additional dose to the workers. The inspectors concluded that the work planning and execution issues that were within the licensee's ability to foresee and correct, and therefore that should have been prevented, resulted in collective doses greater than 5 Rem and greater than 150 percent of the initial dose estimate.

The licensee entered this issue into their CAP as CR-PLP-2012-05812, UT Exams of the Additional CRD Stalk Housings Has Exceeded the Dose Estimate for the RWP. Corrective actions were implemented to address the outage planning and work execution issues.

<u>Analysis</u>: The failure to appropriately incorporate ALARA strategies and insights while planning and executing CRDM 24 repairs during an August 2012 maintenance outage was a performance deficiency that warranted a significance evaluation.

The inspectors determined that the finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because the finding was associated with the Program and Process attribute of the Occupational Radiation Safety cornerstone and adversely impacted the cornerstone objective of ensuring the adequate protection of worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation. Additionally, the finding was similar to IMC 0612, Appendix E, Example 6.i.

The inspectors screened this finding in accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process." The inspectors determined that the finding did not involve: (1) a radiological overexposure; (2) a substantial potential for an overexposure; or (3) a compromised ability to assess dose. The inspectors also determined that the finding involved ALARA planning and work controls and that the licensee's 3-year rolling collective dose average was above 135 person-Rem at the time the performance deficiency occurred. However, because the work activity was a single occurrence that involved an actual dose outcome that was within the licensee's control of less than 25 person-Rem, this finding was determined to be of very low safety significance (Green).

This finding had an associated cross-cutting aspect in the Work Management (H.5) component of the Human Performance cross-cutting area because the work process included the identification and management of risk commensurate to the work and the need for coordination with different groups or job activities.

<u>Enforcement</u>: This finding did not involve enforcement action because no violation of a regulatory requirement was identified. The licensee entered this issue into their CAP as

CR-PLP-2012-05812, UT Exams of the Additional CRD Stalk Housings Has Exceeded the Dose Estimate for the RWP. Corrective actions were implemented to address the outage planning and work execution issues. URI 05000255/2013005-04 is closed. **(FIN 05000255/2014002-06, Failure to Maintain Radiation Exposure ALARA During CRDM 24 Repairs)**

.2 Radiation Worker Performance (02.05)

a. Inspection Scope

The inspectors observed radiation worker and radiation protection technician performance during work activities being performed in radiation areas, airborne radioactivity areas, and high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers were familiar with the work activity scope and tools to be used, workers used ALARA low-dose waiting areas) and whether there were any procedural compliance issues (e.g., workers were not complying with work activity controls). The inspectors observed radiation worker performance to assess whether their training and skill level was sufficient for the radiological hazards and work involved.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications (IE04) performance indicator (PI) for the period from January 1, 2013, through December 31, 2013. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in Nuclear Energy Institute (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, CRs, and NRC Integrated Inspection Reports for the period of January 1, 2013, through December 31, 2013, to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s CR 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.

This inspection constituted one unplanned scrams with complications sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Heat Removal System (MS08) PI for the period from January 1, 2013, through December 31, 2013. 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, CRs, event reports, MSPI derivation reports, and NRC Integrated Inspection Reports for the period of January 1, 2013, through December 31, 2013, 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, whether the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's CR 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.

This inspection constituted one MSPI heat removal system sample as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.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 occurrence 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.

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 CR 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 Selected Issue Follow-Up Inspection: Passive Component Failure Review for SW System

a. Inspection Scope

During RFO 1R23, the inspectors reviewed and observed work in the field to address long-standing issues with the SW system. Specifically, the inspectors reviewed work packages, engineering changes, and non-destructive examination testing data for repairs completed on existing leaks in the SW system and inspections conducted while portions of critical and non-critical system piping were open.

The licensee repaired four existing pinhole leaks within the SW system. Three of those were on ASME Class 3 valves and piping in critical portions of the system and one was on a non-critical pipe in the system. One of the critical piping pinhole leaks was identified on an elbow section of piping downstream of a temperature control valve on the outlet side of a CCW heat exchanger. Cavitation-induced erosion was identified inside the elbow area, which was anticipated based on the configuration of the piping and location downstream of a throttled valve. Similar issues had previously occurred onsite. Another SW piping pinhole leak was identified in a flanged area of branched (tee) carbon steel piping downstream of the 1-1 EDG and left train control room HVAC chiller SW supply isolation valve. This branch connection and flange was originally installed as a temporary water supply connection point in 1995, but was never used. The suspected cause of the pinhole leak was biofouling or microbiologically induced corrosion (MIC) due to it being a stagnant flow section of the piping. This branched section was replaced with a straight section of piping. The final pinhole leak repaired on the critical part of the system was in the valve body of MV-SW135, a 4-inch isolation valve on the bypass line of the discharge piping for the 'A' CCW heat exchanger. This valve was downstream of a throttled valve and the cause of the valve body degradation was identified to be cavitation-induced erosion. This valve was replaced with a stainless steel globe valve that was expected to be less susceptible to cavitation-induced erosion.

Inspections were performed when each of these portions of the SW system were opened. No additional MIC/biofouling concerns were identified in these portions of the system. Minor rust and scaling were identified downstream of the elbow that was replaced, but no additional erosion was identified. Finally, the downstream piping from MV-SW135 had indications of erosion, and that section of piping was replaced along with the valve.

The SW system was identified as a Top Ten Equipment Issue on site and was also being reviewed as part of the licensee's Passive Component Program. Systems in this program received enhanced licensee scrutiny and oversight of the corrective actions to address identified issues. The engineering department had risk-ranked the SW system piping segments and components to develop replacement and inspection plans for susceptible areas. The licensee planned to replace piping and components with materials less susceptible to cavitation-induced erosion or MIC/biofouling based on industry operating experience with non-destructive examinations or inspections of opportunity as portions of the piping were opened. These plans had begun to be implemented at the end of the inspection period and were planned to continue until all replacements and/or inspections had been completed.

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

b. Findings

No findings were identified.

.4 Selected Issue Follow-up Inspection: CRDM Housing Inspection and Replacement,

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The source of the leakage was determined to be a crack in CRDM 24. The licensee performed an extent of condition examination on eight additional CRDM housings. An evaluation to determine the cause of the cracking was also discussed in CR-PLP-2012-05623, Steam Leak Found on CRD-24. Subsequent to the completion of the root cause evaluation, the NRC performed an inspection to review the root cause report and verify the licensee had adequately assessed the issue and the proposed corrective actions were adequate to prevent recurrence. The results of this inspection were documented in NRC Inspection Report 05000255/2013-002 (ML13134A329). One of the proposed corrective actions was to perform additional extent of condition examinations during the next RFO to determine if the condition identified in CRDM 24 existed in other housings. This extent of condition review consisted of performing additional non-destructive examinations on a sample of CRDM housings that were selected based on risk criteria established by the licensee. In particular, the licensee performed eddy current examinations from the inside surface of the housings in the area affected in CRDM 24 as well as two other welds within the CRDM housings. The licensee planned to address any indications that were identified in accordance with ASME Section XI.

Prior to RFO 1R23, the licensee developed special tooling to perform the eddy current exams and qualified the technique using a mock-up pipe designed and constructed to mimic the currently installed CRDM housings. The inspectors observed the qualification process to ensure it met the ASME Code requirements and was adequate to detect flaws in the CRDM housings.

From January 21 through March 7, 2014, the inspectors completed one inspection sample regarding problem identification and resolution based upon a review of the licensee's corrective actions to prevent recurrence of the CRDM leakage identified in 2012, and as described in CR-PLP-2012-05623. Specifically, the inspectors reviewed the procedures the licensee used to perform the eddy current examination to ensure ASME Code requirements were met. The inspectors also observed the eddy current examinations performed on the CRDM housings, reviewed the qualification records of the individuals performing the examination and analyzing the data, and inspected the licensee's actions in response to the results of the eddy current examinations.

The inspectors reviewed the licensee's actions in accordance with performance attributes identified in IP 71152. Specifically, the inspectors reviewed licensee corrective action records to determine whether: (1) the problems were accurately identified; (2) operability and reportability were adequately ascertained; (3) extent of condition and generic implications were appropriately addressed; (4) classification and prioritization of the problem were commensurate with safety significance; (5) root and contributing causes were identified; (6) corrective actions were appropriately focused to correct the problem; and (7) timely corrective actions were completed or proposed commensurate with the safety significance of the issues.

b. Observations and Conclusions

Based on the identification of indications on the selected sample of CRDM housings to be inspected, the licensee expanded their scope and performed eddy current examinations on all 45 CRDM housings of which 17 CRDM housings were found to have rejectable flaw indications. All the indications were contained within the region surrounding the weld affected in CRDM 24. The inspectors that were onsite to observe the examinations followed the issue closely and engaged the licensee in various discussions to assess whether the actions taken in response to the discovery of these indications were appropriate. These discussions addressed the extent of the indications, what additional examinations and evaluations would be performed, and what corrective actions would be taken to address the issue.

The licensee shipped three CRDM housings to a contract laboratory facility for additional non-destructive and destructive examinations. The selection of these housings was based on the number of indications in the housings as well as previous inspection data for the housings being available. A regional inspector and a technical expert from NRC headquarters were onsite at the laboratory facility to observe the examinations and independently assess what the implications of the results were. Specifically, the inspectors questioned whether any of the flaws identified were potentially through-wall and whether the characteristics of these indications were similar to those identified in 2012. Based on the initial laboratory results as well as leakage monitoring performed on site, it was determined that there were no through-wall flaws identified and the structural integrity of the CRDM housings was not compromised.

Based on the number of indications identified, the licensee replaced all CRDM housings with a design that would eliminate the affected weld and therefore reduce the vulnerability to cracking in this area. The inspectors reviewed the new design to ensure all identified vulnerabilities were adequately addressed and the CRDM housings were constructed in accordance with the applicable codes and standards. The inspectors also verified that the installation and post-installation tests that were performed were completed in accordance with the ASME Code and the licensee's quality assurance program.

The inspectors concluded that based on the replacement of all CRDM housings with the new design and an adequate understanding of the degradation mechanism these CRDM housings were exposed to, a safety concern associated with the operation of the plant with the new CRDM housings did not exist.

c. Findings

No findings were identified.

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

.1 Event Notification (EN) 49773, Indications Identified on CRDM Housings

a. Inspection Scope

On January 29, 2014, the licensee submitted an 8 hour non-emergency Event Notification (EN 49773) due to the discovery of indications in 17 of the 45 CRDM housings that were outside the acceptance criteria delineated in ASME Code, Section XI, IWB-3600, "Analytical Evaluation of Flaws." There was no evidence of through-wall leakage. All CRDM housings were inspected. The inspections were being conducted on the housings as part of an extent of condition review based on through-wall leakage that was identified on a CRDM housing in 2012. Site, regional, and headquarters inspectors reviewed the event report to determine the timeliness of the report. Regional inspectors were on site at the time reviewing the inspections on the CRDMs. Subsequently, the licensee replaced 44 of the 45 housings during the outage (one CRDM housing had been replaced in 2012). All of the CRDM housings (including the one replaced in 2012) incorporated a design change in an effort to eliminate the cause of the cracking.

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

b. Findings

No findings were identified.

.2 Unexpected Continuous Air Monitor Alarm

a. Inspection Scope

Regional health physics inspectors and resident inspectors reviewed the plant's response to unplanned changes in airborne radioactivity levels inside the containment building on January 31, 2014. The inspectors evaluated whether the response complied with station procedures, reviewed whether alpha contamination was adequately considered, and assessed the results of the work in the area.

This follow-up of events inspection constituted one sample as defined in IP 71153-05.

b. Findings

No findings were identified.

4OA5 Other Activities

.1 Conversion of 2013 Cross-Cutting Aspects

The table below provides a cross-reference from the third and fourth quarter 2013 findings and associated cross-cutting aspects to the new cross-cutting aspects resulting from the common language initiative. These aspects, and any others identified since January 2014, will be evaluated for cross-cutting themes and potential substantive cross-cutting issues in accordance with IMC 0305 starting with the 2014 mid-cycle assessment review.

Finding	Old Cross-Cutting Aspect	New Cross-Cutting Aspect
05000255/2013004-01	P.1(d)	P.3
05000255/2013005-01	H.4(b)	H.8
05000255/2013005-02	H.4(b)	H.8
05000255/2013005-05	P.1(a)	P.1

.2 (Closed) URI 05000255/2013005-03: Evaluation of High Radiation Area Controls on the Refuel Floor

This URI was opened in the fourth quarter of 2013 when the inspectors reviewed an event where the licensee failed to implement effective high radiation area controls on April 18, 2012, while work was being performed on the refuel floor. Also in the fourth quarter of 2013, the inspectors opened and closed an NCV for the failure to implement high radiation area controls on two other occasions and locations (NCV 05000255/2013005-02; ML14043A507). The inspectors reviewed the information provided by the licensee regarding the April 18, 2012, event on the refuel floor and determined that this represented another example of the previously documented NCV for inadequate control of entry into high radiation areas. This URI is closed to NCV 05000255/2013005-02 (ML14043A507).

4OA6 Management Meetings

.1 Exit Meeting Summary

- On April 11, 2014, the inspectors presented the inspection results to Mr. A. Vitale 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 Exits Meetings

Interim exits meetings were conducted for:

- The inspection results for the areas of radiological hazard assessment and exposure controls and occupational ALARA planning and controls with Mr. A. Vitale, on January 24, 2014; and
- The results of the inservice inspection with Mr. A. Vitale on March 31, 2014.

The inspectors confirmed that any proprietary information that was no longer being reviewed was returned or destroyed.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Dotson, Regulatory Affairs
G. Katt, System Engineering
J. Milliken, Engineering Supervisor
G. Sturm, ALARA Specialist
D. Watkins, Radiation Protection Manager

Nuclear Regulatory Commission

Eric Duncan, Chief, Reactor Projects Branch 3

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000255/2014002-01	NCV	Inadequate Installation of Steam Generator Nozzle Dams (Section 1R04)
05000255/2014002-02	NCV	Failure to Complete Volumetric Examinations for DM Butt Welds in Branch Connections (Section 1R08.5)
05000255/2014002-03	URI	Spent Fuel Pool Region II Criticality Analysis (Section 1R15)
05000255/2014002-04	NCV	Introduction of Foreign Material Into the SW System (Section 1R20)
05000255/2014002-05	NCV	Failure to Follow Procedures During Reactor Vessel Head Lift (Section 1R20)
05000255/2014002-06	FIN	Failure to Maintain Radiation Exposure ALARA on CRDM 24 Repairs (Section 2RS2.1)

Closed

05000255/2014002-01	NCV	Inadequate Installation of Steam Generator Nozzle Dams (Section 1R04)
05000255/2014002-02	NCV	Failure to Complete Volumetric Examinations for DM Butt Welds in Branch Connections (Section R08.5)
05000255/2014002-04	NCV	Introduction of Foreign Material Into the SW System (Section 1R20)
05000255/2014002-05	NCV	Failure to Follow Procedures During Reactor Vessel Head Lift (Section 1R20)
05000255/2014002-06	FIN	Failure to Maintain Radiation Exposure ALARA During CRDM 24 Repairs (Section 2RS2)
05000255/2013005-04	URI	Evaluation of Dose Received by Workers Repairing CRDM 24 (Section 2RS2.1)
05000255/2013005-03	URI	Evaluation of High Radiation Area Controls on the Refuel Floor (Section 4OA5.2)

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.

<u>1R04 Equipment Alignment</u>

- CR-PLP-2014-0711, Red Danger Tag on CV-0823, January 27, 2014
- CR-PLP-2014-0760, Noted Nozzle Dam Low Air Pressure Alarms, January 28, 2014
- CR-PLP-2014-0761, During Loss of Nozzle Dam Air Pressure the Backup Air Supply System Did Not Control Air Pressure as Expected, January 28, 2014
- CR-PLP-2014-0770, Steam Generator Nozzle Dam Supply Air Routed from Different than Expected Air Source, January 28, 2014
- EN-MA-125, Troubleshooting Control of Maintenance Activities, Revision 16
- M-203, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet 2, Revision 25
- M-204, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet 1A, Revision 43
- M-204, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet A, Revision 8
- M-204, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet 1, Revision 84
- M-208, System Diagram for Service Water System, Sheet A, Revision A
- RFL-SG-2, "S/G Primary Nozzle Dam Installation and Removal," Revision 7
- SOP-15 Attachment 10, Frazil Ice Prevention/Mitigation, Revision 56
- SOP-15, Service Water System, Revision 56
- SOP-3, Attachment 10, Checklist 3.1: Engineered Safeguards System Checklist (Shutdown Cooling Inservice), Revision 94
- SOP-3, Attachment 18, Checklist 3.9: Engineered Safeguards Administrative Control Verification, Revision 93
- Work Order 51627641, E-50B Install Nozzle Dams, January 24, 2014
- Work Order 52436085, Set-up and Remove S/G Air Compressors and Associated Equipment, January 21, 2014

<u>1R05 Fire Protection</u>

- CR-PLP-2014-00466, Fire Brigade Had to Remove Cable on CO2 Extinguisher During Fire Response, January 22, 2014
- CR-PLP-2014-0374, Fire Protection Walkdown of Level 1 Areas Observed Transient Combustible Materials in Fire Area 13 that Challenges the Weight Allowed by Transient Combustible Evaluation 13-067, January 20, 2014
- EN-DC-161, "Control of Combustibles," Revision 9
- Fire Hazards Analysis, Revision 7
- FPIP-2, Fire Emergency Responsibility and Response, Revision 17
- FPIP-4, Fire Protection Systems and Fire Protection Equipment, Revision 31
- Pre-Fire Plan for Auxiliary Building – General Areas/Elevation 590', Fire Area 13
- Pre-Fire Plan for Cable Spreading Room and 1-D Switchgear, Fire Areas 2 & 3

- Pre-Fire Plan for Turbine Building General Areas/Elevation 590', 607', 612', and 625', Fire Area 23
- Pre-Fire Plan for Turbine Lube Oil Room/Elevation 590', Fire Area 22

1R07 Heat Sink Performance

- CCS-M-2, Component Cooling Water Heat Exchanger Maintenance, Revision 24
- CR-PLP-2014-0792, Denting Identified on 3 Tubes During E-54A Eddy Current Inspections, January 29, 2014
- CR-PLP-2014-0827, Very Slow Leaks Identified in Six Welded Tube Plugs of E-54A, January 30, 2014
- CR-PLP-2014-0834, Eddy Current Identified Two Flow Tubes in E-54A that Exceed 70% Wall Thinning Criteria, January 30, 2014
- CR-PLP-2014-0903, Leak Identified on Tube Sheet Column 1 Row 6, February 1, 2014
- CR-PLP-2014-1001, After Pressurizing CCW Heat Exchanger with CCW Water, 4 Tube Locations Were Identified as Leaking, February 3, 2014
- CR-PLP-2014-1016, Three More Leaks Identified While Performing Work on CCW Heat Exchanger, February 4, 2014
- CR-PLP-2014-1024, Discovered 3 Tube Sheet Leaks on E-54A, February 4, 2014
- CR-PLP-2014-1294, 3 More Leaking Welded Plugs Identified on E-54A, February 11, 2014
- CR-PLP-2014-1325, Two Existing Welded Plugs Were Found Leaking on E-54A and No Condition Report was Written at the Time, February 13, 2014
- CR-PLP-2014-1367, During Return to Service, Ops Identified North End Bell Leaking on 'A' CCW Heat Exchanger, February 14, 2014
- CR-PLP-2014-1403, During VT-2 Found Small Leak on Bottom of Bolted Connection on North End Bell of E-54A, February 15, 2014
- EC No. 42743, Provide Design and installation Instructions for Mechanically Driven Tubesheet Plug for Component Cooling Water Heat Exchangers, E-54A and E-54B, Revision 0
- Palisades Nuclear Plant Component Cooling Water Heat Exchanger E-54A Inspection Report, January 2014
- Work Order 52473708, E-54A CCW Heat Exchanger Inspection, Cleaning, and Tube Plugging, February 4, 2014
- EA-TWK-95-01; Increase CCW HX Tube Wall Loss Plugging Criteria to 70%; Revision 0
- ENO-17-PN1-01, Final Inspection Report CCW-HX-E-54A, February 19, 2013
- Anatec Final Inspection Report CCW-HX-E-54A, January 2014

1R08 Inservice Inspection Activities

- 12-PAL-0059, F-9, Boric Acid Filter has Boric Acid Buildup Coming from Under Insulation, May 17, 2012
- 12-PAL-0086, Boric Acid Discovered on MO-3081 "HPSI to Cold Leg-Hot Leg INJ Mode Select Packing Area," May 21, 2012
- 13-PAL-0018, CV-1059, Pressurizer Spray Valve from Loop 2A has an Excessive Packing Leak, July 2, 2013
- 13-PAL-003, P-66B High Pressure, Safety Injection Pump Boric Acid Evaluation, May 10, 2013
- 13-PAL-016, P-67B High Pressure, Safety Injection Pump Boric Acid Evaluation, May 22, 2013
- 54-ISI-494-000, Multi-Frequency Eddy Current Array Probe Examination of Ventline and RVLIS Nozzle Bores, August 8, 2012

- 54-ISI-604-012, Automated Ultrasonic Examination of Open Tube RPV Closure Head Penetrations, September 3, 2013
- CC N-729-1, Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1, March 28, 2006
- CC N-770-1, Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds
- CEP-NDE-0640, Non-Section XI Liquid Penetrant Examination (PT), Revision 9
- CEP-NDE-0641, Liquid Penetrant Examination (PT) for ASME Section XI, Revision 7
- CEP-NDE-0901, VT-1 Examination, Revision 4
- CEP-NDE-0902, VT-2 Examination, Revision 7
- CEP-NDE-0955, VT Exam of Bare Metal Surfaces, Revision 303
- CR-PLP-2012-03018, VT-3 Exam Rejected Due to 10% Delta Between Indicator Readings and Load Settings, April 21, 2012
- CR-PLP-2012-03153, Late Notification of a Relevant Condition Associated with a 1R22 Inservice Inspection Examination, April 24, 2012
- CR-PLP-2012-03164, Rejectable Indication on PCS-2-LDL-2B1-1, April 24, 2012
- CR-PLP-2012-03165, Relevant Indication on PCS-4-PRS-1PP1-1, April 24, 2012
- CR-PLP-2012-03344, Bare Metal Visual Exam of Reactor Pressure Vessel Head, April 28, 2012
- CR-PLP-2012-05825, Boric Acid Deposits from CRD-24, August 21, 2012
- CR-PLP-2013-00687, FME Discovered During Mode 3 Walkdown in Containment, February 16, 2013
- CR-PLP-2013-02600, FME Cover Falls into SIRW Tank Nozzle, June 12, 2013
- CR-PLP-2014-00473, Rejectable Linear Indication on Weld PCS-2-LDL-2B1-10A, January 22, 2014
- CR-PLP-2014-00706, Steam Generator E-50A and E-50B Eddy Current Analysis has Indicated Tubes Requiring Repair by Tube Plugging, January 27, 2014
- CR-PLP-2014-00777, Examiner Failed to Perform Illumination Check on VT-3 Examination, January 28, 2014
- CR-PLP-2014-00864, Plug at Location R24C11 Extends Beyond the End of the Tube Sheet, January 30, 2014
- CR-PLP-2014-01073, UT of PCS-6-PRS-iCi-1 (RV-1041) Revealed Two Axial Flaws, February 5, 2014
- EN-DC-319, Boric Acid Corrosion Control Program (BACCP), Revision 10
- NDE Report No. 1R22-PT-12-014, PT of Nozzle-to-Pipe Weld, PCS-2-DRL-1B1-1, April 15, 2012
- NDE Report No. 1R22-VE-12-002, UT of Nozzle-to-Pipe Weld on PCS-2-LDL-2B1-1
- NDE Report No. 1R23-UT-14-007, UT of Pipe-to-Elbow Weld FWS-4-AWS-1S1-250, February 1, 2014
- NDE Report No. 1R23-VT-14-069, VT-3 of Pipe Restraint, CVC-2-LDL-2B2-21PR(H-1.7), January 28, 2014
- NDE Report No. 1R23-VT-14-070, VT-3 of Pipe Restraint, ESS-12-SIS-1LP-233PR(H713), January 28, 2014
- PQR 107, for WPS-SS-8/8-B, GTAW, P-8 to P-8, Revision1
- PQR 170, for WPS-SS-8/8-B, GTAW, P-8 to P-8, Revision 1
- PQR NWS-PQR-04, PQR for WPS NWS-WPS-04, March 17, 2009
- WO 55382551, Replace Valve 2-CTS-160W, April 2, 2012
- WPS NWS-WPS-04, Manual Gas Tungsten Arc Welding, P-8 to P-1, Revision 6
- WPS-SS-8/8-B, Manual Gas Tungsten Arc Welding, P-8 to P-8, Revision 0

1R11 Licensed Operator Requalification Program

- AOP-25, "Loss of Refueling Water Accident," Revision 0
- EN-OP-115, "Conduct of Operations," Revision 14
- EN-OP-116, "Infrequently Performed Tests or Evolutions," Revision 12
- EN-RE-327, "PWR Startup Critical Predictions and Evaluation Process," Revision 3
- GOP-11, "Refueling Operations and Fuel Handling," Revision 47
- GOP-14, "Shutdown Cooling Operations," Revision 46
- GOP-3, "Mode 3 ≥525°F to Mode 2," Revision 31
- PL-LOR-JIT-001S, "Critical Approach by Dilution," Revision 10, March 5, 2014
- SOP-1A, "Primary Coolant System," Revision 22

1R12 Maintenance Effectiveness

- CAP040173, PPAC RWS209 has Questionable Test Conditions and Lack Acceptance Criteria, February 24, 2004
- CR-PLP-2009-00043, Maintenance Rule (a)(1) Action Plan for Charging Pumps P-55A, P-55B, and P-55C, Revision 11
- CR-PLP-2011-04641, No Charging Flow When Restarted P-55A, September 17, 2011
- CR-PLP-2011-04827, Charging Pump P-55B Suction Relief Valve RV-2096 Lifted and Did Not Reseat, September 25, 2011
- CR-PLP-2011-06714, Chemical Volume Control Relief Valves RV-2006 and RV-2082 are in Maintenance Rule (a)(1), December 8, 2011
- CR-PLP-2012-03042, Charging Pump P-55C Did Not Start, April 22, 2012
- CR-PLP-2012-05123, Charging Pump P-55C Degraded Flow, July 17, 2012
- CR-PLP-2012-07051, P-55C Started and then Immediately Tripped, November 4, 2012
- CR-PLP-2013-02920, P-55B Discharge Flush Manifold Weld Failure, July 3, 2013
- CR-PLP-2013-04372, P-55B Discharge Flush Manifold Weld Failure, October 7, 2013
- CR-PLP-2013-04385, P-55A Tripped After Starting, October 7, 2013
- CR-PLP-2013-04596, (a)(1) Evaluation of the Charging Pumps P-55A/B/C and the Chemical and Volume Control System, December 18, 2013
- CR-PLP-2013-04596, P-55A Tripped After Starting, October 24, 2013
- CR-PLP-2013-04687, Charging Pump P-55B Suction Accumulator Pressure High, October 30, 2013
- CR-PLP-2013-04747, Charging Pump P-55A Suction Accumulator Pressure High, November 4, 2013
- CR-PLP-2013-04772, Charging Pumps P-55B and P-55A have Two Functional Failures, November 5, 2013
- CR-PLP-2013-04838, Charging Pump P-55C Suction Accumulator Pressure High, November 8, 2013
- CR-PLP-2013-05039, P-55A Tripped After Starting, November 26, 2013
- DBD-7.08, Plant Protection Against Flooding, Revision 6
- EN-DC-150, Condition Monitoring of Maintenance Rule Structures, Revision 6
- EN-DC-203, Maintenance Rule Program, Revision 2
- EN-DC-206, Maintenance Rule (a)(1) Process, Revision 3
- MSM-M-16, Inspection of Watertight Barriers, January 13, 2011
- Palisades Maintenance Rule Functional Failures, Component Failures and Reliability Events, January 2009 through July 2013, Chemical and Volume Control
- Palisades Maintenance Rule Performance Indicators, Chemical and Volume Control
- PMRQ 50081032, CK-FW419, Inspection Per EM-28-02, March 24, 2014
- PMRQ 50083334, P-55B Accumulator PM

- PMRQ 50083335, P-55C Accumulator PM
- PMRQ 50083363, P-55A Accumulator PM
- PMRQ 50084248, CK-RW421, Inspect PM Per EM 28-02*, March 24, 2014
- SEP-CV-PLP-001, Applicable Check Valve Program Components, Attachment 7, Revision 2
- WO 00213227 01, CK-RW421, Inspect PM Per EM 28-02*, November 3, 2010
- WO 51627757 01, CK-RW421, Inspect PM Per EM 28-02*, March 27, 2014
- WO 51627757 01, CK-RW421, Inspect PM Per EM 28-02*, March 27, 2014
- WO 51626037, P-55A Accumulator Pressure Test
- WO 51626039, P-55B Accumulator Pressure Test
- WO 51626040, P-55C Accumulator Pressure Test

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>

- CR-PLP-2014-0399, While Lifting West Side Crossover Pipe of Low Pressure Turbine B, the Turbine Building Crane Main Hoist Stopped Responding to Raise/Lower Signals from Remote Control, January 21, 2014
- CR-PLP-2014-0421, While Lifting East Side Crossover Pipe of Low Pressure Turbine B, the Turbine Building Crane Main Hoist Stopped Responding to Raise/Lower Signals from Remote Control, January 21, 2014
- CR-PLP-2014-0432, Identified Insufficient Rating for Rigging Used to Lift Reactor Head Lift Rig Before Actual Lift Occurred, January 22, 2014
- CR-PLP-2014-0647, Turbine Crane Lifting Loads Over Workers, January 25, 2014
- CR-PLP-2014-0715, While Attempting to Install Bladder into Service Water Header, Vacuum was Too Great and Ripped Bladder Off of Lanyard and Went Down Into the Piping, January 27, 2014
- CR-PLP-2014-0716, Cannot Replace MV-SW135 and Piping Downstream of CV-0823 Due to Isolation Issues at this Time, January 27, 2014
- CR-PLP-2014-0809, Water Coming Out of MV-SW135 Due to Installed Bladder Deflating During Maintenance, January 29, 2014
- CR-PLP-2014-0912, Rotary Count Limit Switch on L-2, Turbine Building Crane, Does Not Support Lifting of Turbine Loads, February 1, 2014
- CR-PLP-2014-1228, Operator Left Load on Hook to Move to Low Dose Area for 2 Hours in Containment, February 9, 2014
- CR-PLP-2014-1865, Cylinder 9L Fuel Pump Control Rack Latch Both Backed Out Approximately 0.25 Inches from Tight on 1-1 EDG, March 4, 2014
- CR-PLP-2014-1867, During Monthly Testing, 1-1 EDG, Would Not Stop When Local Engine Control Switch Taken to Off, March 4, 2014
- CR-PLP-2014-1893, Load Flown Over Workers in Reactor Cavity, March 4, 2014
- Drawing E-8, Sheet 1, 125VDC, 120V AC Instrument and Preferred AC System, Revision 57
- Drawing E-8, Sheet 2, 125VDC, 120V AC Instrument and Preferred AC System, Revision 55
- EPS-E-10A, ED-11-1 Breaker Inspection and Testing, Revision 0
- FHS-M-24, Movement of Heavy Loads in Containment Building, Revision 38
- MSM-M-72, Movement of Heavy Loads in the Turbine Building, Revision 1
- Operations Contingency Plan for 1C Bus Outage
- PE-P7522, Operating and Maintenance Instructions for the Cold Leg Plugs for Palisades Station, Revision 1
- RFL-D-16, Reactor Vessel Closure Head Removal, Revision 14
- SOP-30 Checklist 30, Station Power System Checklist, Revision 74
- WO 212771, 72-307 DC Breaker Replacement
- WO 212772, 72-308 DC Breaker Replacement
- WO 324812, CV-0823, Critical Service Water Pipe Leak Downstream CCW HX

- WO 52435688, ED-11-1
- WO 353022, N-50 Cold Leg Nozzle Plug Installation and Removal, February 12, 2014
- WO 376223, K-6A, 1-1 Emergency Diesel Generator, Failed to Shutdown, March 4, 2014

1R15 Operability Determinations

- CR-PLP-2014-00086, Relay Manufacturers were Changed for a Reed Relay, January 7, 2014
- EC 43397, Acceptability of the Installed Static Switch Control Boards in the Inverters, January 14, 2014
- EN-CS-S-012L, Attachment 4, Seismic Adequacy Verification Checklist for Replacement Parts, Omron MY4 and MY4N Relays, January 13, 2014
- EN- CS-S-012L, Attachment 4, Seismic Adequacy Verification Checklist for Replacement Parts, Computer Components 300-470A Relay, January 13, 2014
- EN-CS-S-012L, Control of Seismic Qualification Utility Group (SQUG) Generic Implementation Procedure (GIP) (EM-18-08), Revision 0
- CR-PLP-2013-4775, Issues Identified with Region II of Spent Fuel Pool Criticality Analysis, November 5, 2013
- CR-PLP-2014-0127, Received Diesel Gen. 1-1 Fail to Start Alarm During Overspeed Trip Setpoint Verification Testing, January 9, 2014
- Equipment Apparent Cause Evaluation: K-6A; Failed to Start During Overspeed Trip Testing, Revision 0
- Work Order #371600, Troubleshooting for K-6A Failure to Start During Overspeed Trip Testing, January 9, 2014
- Work Order #52434584, Inspect BF and BFD Relays in EC-20, July 1, 2013
- Work Order #345493, Boroscope 1-2 EDG Relay Contacts in Panel EC-30, March 25, 2103
- Work Order #349218, Replace Cable Labels Inside Panel EC-20, March 21, 2014
- Work Order #349219, Replace Cable Labels Inside Panel EC-30, March 21, 2014
- Lower Tier Apparent Cause Evaluation Report: K-6B, Emergency Diesel Generator Failed to Start During MO-7A-2 While Using ASM-2A, Diesel Generator Air Start Motor, Revision 1
- VEN-M12, Schematic Diagram Engine Control DG 1-1, Sheet 98 (1), Revision 36
- CR-PLP-2014-00042, Additional K-Line Breakers Identified Subject to Part 21 from ABB, January 3, 2014
- CR-PLP-2014-00022, ABB Part 21 Received by Palisades for 480V K-Line breakers, January 2, 2014
- FSAR Chapter 8, Electrical Systems, Revision 30
- FSAR Chapter 7, Instrumentation and Controls, Revision 27
- ABB Letter dated November 25, 2013, 10 CFR Part 21 Notification of Deviation Re: K-Line Circuit Breaker Primary Close Latch

1R19 Post Maintenance Testing

- COP-22A, Diesel Fuel Oil Sampling Program, Revision 20
- CR-PLP-2014-01437, Longer Prime Times for P-18A/B, February 16, 2014
- CR-PLP-2014-01569, Extractor Valve Found Installed Incorrectly during Troubleshooting for P-18A/B, February 18, 2014
- CR-PLP-2014-0819, VC-10, CRHVAC Chiller Tripped, January 29, 2014
- CR-PLP-2014-1392, CV-2155 Gasket ID Too Small for Replacement Activities, February 14, 2014
- CR-PLP-2014-1612, Flange on Upper Oil Cooler of P-50C Motor Found Leaking, February 22, 2014

- CR-PLP-2014-1628, Incorrect Torque Value in Installation Procedure PCS-E-46C for CCW Flange Leak on P-50C Motor, February 22, 2014
- CR-PLP-2014-1635, Flanges on P-50C Motor Were Found Out of Parallel, February 22, 2014
- Drawing M-398, Level Settings Diagram for T-10A, Revision 9
- EA-SC-96-051-01, Test Plan for Original Installation of P-18A/B Fuel Oil Transfer Pumps
- PCS-E-46C, Primary Coolant Pump P-50C Motor Removal and Installation, Revision 7
- PCS-M-53, Removal and Installation N-9000 Series Primary Coolant Pump Shaft Seal Cartridge, Revision 19
- QE-35A, ED-01 Battery Checks – Quarterly, Revision 9
- RE-83A, Service Test – Battery No ED-01, Revision 21
- WI-CVC-M-02, Repairing Boric Acid Blender Outlet Control Valve CV-2155, Revision 0
- WO 374161, MC-17, Fuel Oil Sampling
- WO 309163, ED-01 Replace Cells 11, 39, 55, and 56 in 1R23, January 25, 2014
- WO 332376, Remove and Reinstall Breaker EMA-2104 for P-50C Pump Work, February 21, 2014
- WO 369026, CV-2155 Leaks By Affecting PCS Leakrate, February 13, 2014

1R20 Outage Activities

- AOP-25, "Loss of Refueling Water Accident," Revision 0
- AOP-26, "Loss of Spent Fuel Pool Cooling," Revision 1
- AOP-30, "Loss of Shutdown Cooling," Revision 0
- AOP-34, "Fuel Handling Accident," Revision 0
- BOP-PT-14-003, Liquid Penetrant Examination of Heavy Load Lifting Device, January 24, 2014
- BOP-PT-14-004, Liquid Penetrant Examination of Heavy Load Lifting Device, January 24, 2014
- BOP-PT-14-005, Liquid Penetrant Examination of Core Support Barrel Lift Device, January 13, 2014
- BOP-VT-14-004, Visual Examination of Pipe Hanger, Support or Restraint, (VT-1) Core Support Barrel Lifting Device, January 24, 2014
- CEP-NDE-0255, Radiographic Examination of Welds and Components, Revision 6
- CR-PLP-01405, Undertolerance Dimension on the Flange Lip During Final Machining on CRD-31 Housing, February 15, 2014
- CR-PLP-2013-04197, Gold-Colored, Shiny Deposits in the Drip Pan of VHX-3, September 24, 2013
- CR-PLP-2014-0000856, Visual Inspection of Component ES-GB-T006 Indicated Significant Wall Thinning, January 30, 2014
- CR-PLP-2014-00296, Display Failed for the Wide Range Log Scale Indicator, January 17, 2014
- CR-PLP-2014-00302, Temporary High Lube Oil Differential Pressure Cue to Cold Lube Oil, January 18, 2014
- CR-PLP-2014-00306, An Alloy 600 Project Elbow Weld Assembly Was Not Supplied with an N-2 Form, January 18, 2014
- CR-PLP-2014-00317, Loose Roofing Material on Turbine Building, January 19, 2014
- CR-PLP-2014-00328, Two Unanticipated Electronic Alarming Dosimeters Dose Rate alarms Received During Inspection of CV-1059 and CV-1057, January 20, 2014
- CR-PLP-2014-00370, Chemistry Identified Two Areas of Air Inleakage into the Condensate System that Need to be Addressed, January 20, 2014
- CR-PLP-2014-00441, History of Valve Leakage May Warrant Performance of the RO-112 Attachments as Normal Method of Testing, January 22, 2014

- CR-PLP-2014-00443, Several Items Dropped to Reactor Head While Removing Control Rod Drive Motor Packages, January 22, 2014
- CR-PLP-2014-00468, CV-0910 Failed to Meet Its Close Stroke time, January 22, 2014
- CR-PLP-2014-00469, Damage Sustained by RV-3267 Due to Being Installed Cocked, January 22, 2014
- CR-PLP-2014-00473, Rejectable Linear Indication 0.300 Inches in Length on Elbow Fitting During Dye penetrant Test of ISI Weld PCS-2-LDL-2B1-10A, January 22, 2014
- CR-PLP-2014-00501, "A" Steam Generator Snubbers Require Inspection, January 23, 2014
- CR-PLP-2014-00519, Blown Control Power Fuse FUX/B2525 Due to Jumper Not Secured Properly, January 23, 2014
- CR-PLP-2014-00580, PCS Water Level Rising Due to Transfer from SIRW to Hot Leg, January 24, 2014
- CR-PLP-2014-00610, T-3 CCW Surge Tank level was Lowering at a Rate of Approximately 1% an Hour, January 25, 2014
- CR-PLP-2014-00623, P-18A Fuel Oil Transfer Pump Performing Inadequately, January 25, 2014
- CR-PLP-2014-00643, Inner Bearing Journal Plate on P-1B Main Feedwater Pump, January 25, 2014
- CR-PLP-2014-00650, Significant Spindle Wear Found During Inspection of RV-0701 and NWS Technologies Recommends Replacement as Documented in CEAR 14-26, January 26, 2014
- CR-PLP-2014-00652, Significant Spindle Wear Found During Inspection of RV-0712 and NWS Technologies Recommends Replacement as Documented in CEAR 14-29, January 26, 2014
- CR-PLP-2014-00653, Significant Spindle Wear Found During Inspection of RV-0723 and NWS Technologies Recommends Replacement as Documented in CEAR 14-28, January 26, 2014
- CR-PLP-2014-00654, Cracks Found in RV-0723 Nozzle and NWS Technologies Recommends Replacement as Documented in CEAR 14-30, January 26, 2014
- CR-PLP-2014-00665, Support Tube Located at CRD-17 Became Hung-Up in Upper Housing During Inspection, January 26, 2014
- CR-PLP-2014-00668, Current SB Nozzle Dam Control Consoles Have a Single Point of Failure Issue, January 26, 2014
- CR-PLP-2014-00678, Dead Animal Plugging Many Tube Ends of #-10 West Inlet, January 26, 2014
- CR-PLP-2014-00708, Loose Sheet Metal Panel on Turbine Building, January 27, 2014
- CR-PLP-2014-00755, Discharge Ball Valve on M-96 is Broken, January 28, 2014
- CR-PLP-2014-00816, Fuel Transfer System Received Overload Condition, January 29, 2014
- CR-PLP-2014-00823, Primary System Drain Tank T-74 Stroke Time Exceeded the Maximum, January 30, 2014
- CR-PLP-2014-00853, WO-319183 FAC Replacement Min Wall Issue, January 30, 2014
- CR-PLP-2014-00873, WO for ED-08 (Inverter #3) Overhaul Failed to Include Steps to Fully Incorporate Engineering Change, January 31, 2014
- CR-PLP-2014-00874, Steam Erosion Identified During 1R23 Dismantled Inspection of K-1-LPA and K-1-LPB, January 31, 2014
- CR-PLP-2014-00878, Galls Developed During Disassembly of K-1-LPB, January 28, 2014
- CR-PLP-2014-00882, System Engineer Completed Walkdown of E-30 Cooling Tower, January 31, 2014
- CR-PLP-2014-00885, NRC Identified Need to Evaluate Previously Identified Ultrasonic Examination Indications, January 31, 2014
- CR-PLP-2014-00917, Gouge/Gall Occurred During Drilling Operations, February 1, 2014
- CR-PLP-2014-00968, Steam Cuts on CV-0779, February 3, 2014
- CR-PLP-2014-00969, Steam Cuts on CV-0782, February 3, 2014

- CR-PLP-2014-01038, Foreign Material Observed on Top of the Cycle 23 Core, February 1, 2014
- CR-PLP-2014-01090, Reactor Side Motor Electric Clutch of H-5 Fuel Transfer Carrier Unexpectedly Found Energized, February 5, 2014
- CR-PLP-2014-01125, Two Human Performance Errors Resulting in Failure of AREVA Equipment, February 6, 2014
- CR-PLP-2014-01135, White Staining and Overall Corrosion Identified During Bare Metal Visual Examination of Reactor Pressure Vessel Head, February 7, 2014
- CR-PLP-2014-01138, Restricted Access for RAD Worker for Ducking Under HRA Boundary Without Permission, February 7, 2014
- CR-PLP-2014-01149, NRC Reported Door-59 Found Not Fully Dogged, February 7, 2014
- CR-PLP-2014-01176, FME Bladder Lost During Work Near CV-0823
- CR-PLP-2014-01210, Missed Administrative Requirement of GOP-14 Att. 11 Section 7.3.d, February 9, 2014
- CR-PLP-2014-01246, Worker Received Dose Rate Alarm Working on Stuck Reactor Head Stud, February 10, 2014
- CR-PLP-2014-01254, Six Pieces of Foreign Material Identified in Bottom of Core Barrel, February 10, 2014
- CR-PLP-2014-01254, Six Pieces of Foreign Material Identified in Bottom of Core Barrel, February 10, 2014
- CR-PLP-2014-01256, Individual Received Dose Rate Alarm During Removal of CRDM Cutter Tool, February 11, 2014
- CR-PLP-2014-01263, Worker’s Hand was Unmonitored During Removal of Highly Irradiated Sleeve, February 11, 2014
- CR-PLP-2014-01266, Two Wear Marks in Quadrant Two on Control Rod 110, February 11, 2014
- CR-PLP-2014-01267, Wear Mark in Quadrant Two on Control Rod 204, February 11, 2014
- CR-PLP-2014-01270, Screw Missing from Upper Cover 152-103 SW Pump P-7B, February 11, 2014
- CR-PLP-2014-01276, VC-11, Control Room HVAC Refrig Condensing Unit Tripped and Trouble Light Actuated, February 11, 2014
- CR-PLP-2014-01317, Normal Monthly TS Testing is Not Currently Applicable in Current Mode of Plant Operation, February 12, 2014
- CR-PLP-2014-01320, Temporary Modification Installed Incorrectly on H-6 Reactor Side Tilt Machine, February 12, 2014
- CR-PLP-2014-01330, Equipment Issue with Pump During Installation of the PCS Cold leg Loop Plugs, February 13, 2014
- CR-PLP-2014-01333, Deformation Identified on One Crimping Cup on Upper Tie Plate on Fuel Assembly Z41, February 13, 2014
- CR-PLP-2014-01336, System Differential Pressure Indicating Leakage After Closing Bent Isolation Valve, February 13, 2014
- CR-PLP-2014-01346, Loop Plug at PCS 1B Cold Leg from Primary Coolant Pum P50B Leaking, February 13, 2014
- CR-PLP-2014-01394, CV-3002 Did Not Indicate Full Open During Testing, February 14, 2014
- CR-PLP-2014-01410, Trend Associated with Lifting/Rigging and Material Handling, February 15, 2014
- CR-PLP-2014-01471, Unacceptable Condition with Pipe Restraint PCS-4-PRS-1P3-4PR (H-866.2), February 17, 2014
- CR-PLP-2014-01535, Unacceptable Condition with Pipe Restraint PCS-4-PRS-1P2-2PR (H-875), February 19, 2014

- CR-PLP-2014-01538, Water Flowing from a Swagelock Valve Downstream of MV-PC1178A P50C Seal Flush Filter F-60C Drain During Refill of Primary Coolant System, February 19, 2014
- CR-PLP-2014-01563, Auxiliary Feed Pump P-8C Recirculation Line Documentation Could Not be Located, February 20, 2014
- CR-PLP-2014-01572, Stud Number 2 Threads Rolled Due to Contact with Omega Seal Weld, February 20, 2014
- CR-PLP-2014-01574, Final Machining of CRD-41 Resulted In an Out of Tolerance Condition at the Rx Head Nozzle Flange, February 20, 2014
- CR-PLP-2014-01576, Thermal Sleeves at the Nozzle Locations for CRD-44, CRD-16 and CRD-18 Could Not Be Inserted, February 20, 2014
- CR-PLP-2014-01592, Secondary Position Indication Cables #11, 13, 23, 22 and 43 Were Found Degraded and in Need of Repair, February 21, 2014
- CR-PLP-2014-01630, Trend in Rigging and Lifting Issues, February 22, 2014
- CR-PLP-2014-01636, Work Order 52326018 to Replace Seals on Containment Equipment Hatch is Being Deferred, February 23, 2014
- CR-PLP-2014-01659, Stand Down Needed Regarding Work Group Safety in Containment, February 23, 2014
- CR-PLP-2014-01660, Refueling Monitors Alarm During Core Support Barrel Removal and Repositioning, February 24, 2014
- CR-PLP-2014-01738, Fuel Assemble AA57 Was Not Able to Be Installed in Core Location R-17, February 27, 2014
- CR-PLP-2014-01748, Pinhole Leaks in Service Water Outlet TCV CV-0839, February 27, 2014
- CR-PLP-2014-01778, Significant Bow and/or Twisting in Several Once-Burned Batch AA Fuel Assemblies, February 28, 2014
- CR-PLP-2014-01805, A Section of Service Water Piping from VHX-3 Is Not Installed Properly, March 1, 2014
- CR-PLP-2014-01827, CV-0781, ADV, Does Not Have Open Light Indication, March 2, 2014
- CR-PLP-2014-01852, Position Indicators for CV-0511 (Turbine Bypass to Condenser) Were Not Illuminated During Functional Testing, March 3, 2014
- CR-PLP-2014-01900, Pressurizer Relief Valve RV-1041 Replacement Flange Measurements Did Not Meet Required Design Tolerances, March 5, 2014.
- CR-PLP-2014-01901, Dye Penetrant Examination of Replacement Flange for Pressurizer Relief Valve RV-1041 Identified Four Rejectable Indications, March 5, 2014
- CR-PLP-2014-01955, Minimum Wall and Internal Taper Measurements Appear Out of Tolerance, March 6, 2014
- CR-PLP-2014-01978, Debris Noted on Containment Floor Drains, Marcy 7, 2014
- CR-PLP-2014-02002, Water Becomes Entrained in Temporary Hose for PCS Level Glass, March 9, 2014
- CR-PLP-2014-02005, Degraded Vacuum in the Head Discovered During Vacuum Fill, March 9, 2014
- CR-PLP-2014-02007, At-the-Controls Operated Distracted by Pre-Job Brief, March 9, 2014
- CR-PLP-2014-02008, Reactor Head Vent, MV-PC1060A Locked Open, March 9, 2014
- CR-PLP-2014-02061, Evidence of Leakage Issues During VT-2 or East ESG, March 12, 2014
- CR-PLP-2014-0816, Fuel Transfer Machine Overload Condition During Functional Checks, January 29, 2014
- CR-PLP-2014-0948, Dual Position Noted on Refueling Transfer Carriage, February 2, 2014
- CR-PLP-2014-1020, Transfer Winch on Reactor Side of Fuel Transfer Carriage Trips on Overload, February 4, 2014

- CR-PLP-2014-1027, Refueling Machine Receiving Overload Indications, February 4, 2014
- CR-PLP-2014-1052, Refueling Machine Hoist Box Latch Switch Found Loose, February 5, 2014
- CR-PLP-2014-1548, OCC Directed Radiographers to Perform Second Shot Without Waiting for Results of Test Shot, February 20, 2014
- CR-PLP-2014-1752, Inspection of Containment Sump Envelope Identifies Deficiencies, February 26, 2014
- CR-PLP-2014-1866, Small Debris Noted on Exterior of Containment Sump Passive Strainer Assemblies During RT-92 Inspection, March 4, 2014
- CR-PLP-2014-1881, Reactor Cavity Liner Leakage, March 4, 2014
- CR-PLP-2014-1978, Small Debris Noted on Floor Drain Debris Screens in Containment Sump During Inspection, March 7, 2014
- CR-PLP-2014-2002, Water Became Entrained in Temporary Hose Used for PCS Level Glass During Vacuum Fill, March 9, 2014
- CR-PLP-2014-2005, Less Water than Anticipated was Added to PCS During Vacuum Fill, March 9, 2014
- CR-PLP-2014-2007, NOS Identified NCO ATC Became Involved in Pre-Job Brief During Vacuum Fill Which was a Distraction, March 9, 2014
- CR-PLP-2014-2008, Water Running On Top of Reactor Head Area After Securing from Vacuum Fill, March 9, 2014
- CR-PLP-2-14-00327, CV-3025 Popped Open 30 Minutes After Raising the Output Signal Numerous Times, January 19, 2014
- CR-PLP-2-14-01262, Containment Book Crane Boomed Up Inadvertently During Removal for CRD-44, February 11, 2014
- CR-PLP-2-14-01264, Worker Inadvertently Grabbed Thermal Sleeve During Removal, February 11, 2014
- CR-PLP-2-14-01455, Lower ROSA Arm Dislodged Upper Platform During Removal from the Vessel, February 16, 2014
- CR-PLP-2-14-00897, Airborne Radioactivity Excursion on the 649’ Elevations of Containment Building and Auxiliary Building Spent Fuel Pool, January 31, 2014
- EM-04-29, Guidelines for Preparing Fuel Movement Plans, Revision 13
- EN-DC-115, PCS Vacuum Fill Skid, Revision 11
- EN-DC-136, Provide Recurring Use Temporary Modification for PCS Vacuum Fill, Revision 3
- EN-MA-118, Foreign Material Exclusion, Revision 9
- EN-MA-119, Material Handling Program, Revision 17
- EN-OM-123, Fatigue Management Program, Revision 7
- EN-OP-116, Infrequently Performed Tests or Evolutions, Revision 012
- EN-OU-108, Shutdown Safety Management Program (SSMP), Revision 6
- EN-RE-326, PWR Core Loading Verification, Revision 1
- EN-RE-327, PWR Startup Critical Predictions and Evaluation Process, Revision 3
- EN-RP-150, Radiography and X-Ray Testing, Revision 10
- EOP Supplement 1, Pressure Temperature Limit Curves, Revision 5
- Foreign Material Exclusion Plan, Steam Generators and Pressurizer Manway
- GOP-11, Refueling Operations and Fuel Handling, Revision 47
- GOP-11, Attachment 1, Refueling/Fuel Handling Prerequisite Master Checksheet, Revision 47
- GOP-11, Attachment 5, Containment Closure Valve Alignment, Revision 47
- GOP-14, Shutdown Cooling Operations, Revision 46
- GOP-2, Mode 5 to Mode 3 ≥525°F, Revision 36
- GOP-3, Mode 3 ≥525°F to Mode 2, Revision 31
- GOP-4, Mode 2 to Mode 1, Revision 23

- GOP-9, Mode 3 ≥ 525°F to Mode 4 or Mode 5, Revision 36
- LO-OLPLP-2011-84, Outage Risk Assessment for RO-23, November 12, 2013
- M-201, P&ID Primary Coolant System, Sheet 1, Revision 86
- M-201, P&ID, Primary Coolant System, Sheet 2, Revision 66
- M-202, P&ID Chemical and Volume Control System, Sheet 1A, Revision 63
- M-202, P&ID Chemical and Volume Control System, Sheet 1B, Revision 58
- M-203, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet 2, Revision 25
- M-204, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet 1, Revision 84
- M-204, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet 1A, Revision 43
- MSM-M-71, "Containment Cleanliness Implementation Plan and Containment Closeout," Revision 11
- Palisades Fuel Movement Forms, 1R23 Core Offload, February 5, 2014
- Palisades Fuel Movement Forms, 1R23 Core Reload, February 24, 2014
- PCS-M-50, Detensioning and Tensioning of Pressurizer T-72 Manway, Revision 11
- PMP-RFL-V-8, Removal and Reinstallation of Core Support Barrel, Revision 8
- PO-2, "PCS Heatup/Cooldown Operations," Revision 6
- PO-2, "Specification Surveillance Procedure," Revision 6
- Radiography Shot Plan for Reactor Building Radiography Shots for Alloy 600, Revision 3
- RFL-D-16, "Reactor Vessel Closure Head Removal," Revision 13
- RFL-D-16, "Reactor Vessel Closure Head Removal," Revision 13
- RFL-D-18 Attachment 11, "Remove Incores and RVLMS Detectors from UG, Recommended Tools," Revision 5
- RFL-D-18 Attachment 12, "Remove Incores and RVLMS Detectors from UGS, Incore Detector Configuration," Revision 5
- RFL-D-18 Attachment 13, "Remove Incores and RVLMS Detectors from UGS, "In-Field" Incore Configurations," Revision 5
- RFL-D-18 Attachment 4, "Remove Incores and RVLMS Detectors from UGS, Incore/RVLMS Locking Device Air Manifold Details," Revision 5
- RFL-D-18 Attachment 6, "Remove Incores and RVLMS Detectors from UGS, Incore/RVLMS Tree Assembly and Locking Device Details," Revision 5
- RFL-D-18 Attachment 8, Remove Incores and RVLMS Detectors from UGS, "RVLMS Detector Installation/Removal Tool Details," Revision 5
- RFL-D-18, "Remove Incores and RVLMS Detectors from UGS," Revision 5
- RFL-D-3, "Open Equipment Hatch," Revision 7
- RFL-F-2, "Refueling Machine (RFM)," Revision 9
- RFL-R-13, "Reactor Pressure Vessel Detensioning," Revision 5
- RFL-R-16, "Reactor Vessel Closure Head Installation," Revision 14
- RFL-SG-2, "S/G Primary Nozzle Dam Installation and Removal," Revision 7
- RFL-V-11, "Reactor Vessel Foreign Object Search Reactor Core Map," Revision 2
- RFL-V-7, "Fuel Movement," Revision 13
- RFL-V-9, Core Mapping System Setup and Operation, Revision 5
- RT-191, "Startup Physics Test Program," Revision 9
- RT-92, "Inspection of Containment Sump Envelope," Revision 7
- SOP-1A, "Primary Coolant System," Revision 22
- SOP-1B, "Primary Coolant System – Cooldown," Revision 17
- SOP-1B, "Primary Coolant System – Cooldown," Revision 18
- SOP-1C, "Primary Coolant System – Heatup," Revision 17

- SOP-24, “Ventilation and Air Conditioning System – Fuel Handling Area Ventilation,” Revision 61
- SOP-24, “Ventilation and Air Conditioning System – Ventilation systems Special Alignments,” Revision 61
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- SOP-27, “Spent Fuel Pool Cooling System,” Revision 66
- SOP-28, “Fuel Handling System,” Revision 50
- SOP-2A, “Chemical and Volume Control System,” Revision 78
- SOP-2B, Attachment 2, Checklist 2.1: CVC System Checklist, Revision 46
- SOP-3, “Safety Injection and Shutdown Cooling System,” Revision 94
- SOP-3, “Safety Injection and Shutdown Cooling System,” Revision 93
- SOP-6, “Reactor Control System,” Revision 34
- Tagout MAC-001-EA-11, C-Bus Outage Tasks
- WI-MSM-M-29, Installation and Removal of Primary Coolant System Vacuum Refill Equipment, Revision 4
- WI-PCS-M-06, “NSSS Walkdown,” Revision 4
- WO 324812, CV-0823, Critical Service Water Pipe Leak Downstream CCW HX
- WO 332696-05, Fill and Vent LPSI Piping, Cold Leg Piping Returns to PCS, and Shutdown Cooling Heat Exchangers, E-60A and E-60B, February 16, 2014
- WO 52405416 Task 05, HGR/HC27-H3.5; Coolant Pump P-50B Seal Leakage, April 30, 2013
- WO 52405416-01, M-1011, PCS Vacuum Refilll System, (PCSVRS), February 15, 2012
- WO 52405416-02, PCS Vacuum Refill Equipment (M-1011), February 15, 2012
- WO 52405416-03, PCS Vacuum Refill Equipment (M-1011), February 15, 2012
- WO 52405416-04, M-1011, PCS Vacuum Refill System (PCSVRS), February 13, 2012
- WO 52405416-06, M-1011, PCS Vacuum Refill System (PCSVRS), January 21, 2014
- WO 52436062-01, E-50A, ‘A’ Steam Generator, Remove Primary Manway Covers, January 21, 2014
- WO 52436062-02, E-50A, ‘A’ Steam Generator, Install Nozzle Dams, January 21, 2014

<u>1R22 Surveillance Testing</u>

- CR-PLP-2013-05334, Tave and Power Continuous Rise after Placing T-52 In-Service, December 18, 2013
- CR-PLP-2014-00854, RO-32-69 Local Leak Rate Test Procedure Rendered Invalid Due to Valve Misposition, January 30, 2014
- CR-PLP-2014-0127, During T-302 Control Room Received Diesel Generator 1-1 Fail to Start Alarm, January 9, 2014
- CR-PLP-2014-0138, Replacement of Jacket Water Pressure Relay 2 Due to High Resistance on Contact that Prevented Start of EDG 1-1, January 10, 2014
- CR-PLP-2014-0551, PCV-3057B Did Not Maintain Pressure as Needed for Containment Sump Check Valve Test, January 24, 2014
- CR-PLP-2014-0593, During RO-141, Containment Sump Check Valves Initial Breakaway Torques Exceeded the Breakaway Action Limit, January 24, 2014
- CR-PLP-2014-1099, P-54C Would Not Start During RT-8C Pre-Test Lineup, February 5, 2014
- CR-PLP-2014-1108, Wavebook Failed to Capture Data for Test Starting Time of Diesel Generator 1-1, February 6, 2014
- CR-PLP-2014-111, Observed Unconnected Wires in C-13 Panel During RT-8C, February 6, 2014

- CR-PLP-2014-1434, Service Water Pump P-7A Did Not Meet Acceptance Criteria for Flow During RO-144, February 16, 2014
- CR-PLP-2014-1439, Vibration Analyzer Froze While Taking Outboard Bearing Readings for P-7A, 'A' Service Water Pump During RO-144, February 16, 2014
- CR-PLP-2014-1468, MV-SW342, SW Pump P-7A Mini-Flow, Has a Severity Level 5 Leak, February 17, 2014
- CR-PLP-2014-1587, Post-cal of Instruments for RO-216/T-218 Found 'B' CCW Heat Exchanger Flow Indicator Out of Cal, February 21, 2014
- CR-PLP-2014-1678, P-67A, 'A' LPSI Pump, Had High Differential Pressure During RO-98, February 25, 2014
- CR-PLP-2014-1679, P-54A, 'A' Containment Spray Pump, Had Low Differential Pressure During RO-98, February 25, 2014
- CR-PLP-2014-1839, During RO-65, HPSI Hot Leg Injection Check Valve and Cold Leg/Hot Leg Flow Balance Test, Acceptance Criteria was Not Met, March 2, 2014
- Drawing M-210, Sheet 1A, Clean Radioactive Waste Treatment System, Revision 22
- Drawing M-210, Sheet 1B, Clean Radioactive Waste Treatment System, Revision 19
- DWO-1, Operator's Daily/Weekly Items Modes 1-4, Revision 101
- EN-HU-106, Procedure and Work Instruction Use and Adherence, Revision 2
- EN-MA-143, "Use of Air Operator Valve Diagnostics," Revision 1
- Heat Balance Power Plots, December 18, 2013
- M-201, P&ID Primary Coolant System, Sheet 1, Revision 86
- M-203, P&ID Safety Injection, Containment Spray, and Shutdown Cooling System, Sheet 2, Revision 25
- RO-141, "Containment Sump Check Valves Inservice Test," Revision 5
- RO-144, Comprehensive Pump Test Procedure Service Water Pumps P-7A, P-7B, and P-7C, Revision 10
- RO-216, Service Water Flow Verification, Revision 20
- RO-32-69, Local Leak Rate Test Procedure for Penetration MZ-69, Revision 30
- RO-65, High Pressure Safety Injection (HPSI), Trains 1 and 2, and Hot Leg Injection (HLI) Check Valve Test and Cold Leg/Hot Leg Flow Balance Test, Revision 27
- RO-98, LPSI and Containment Spray Comprehensive Pump Test and Check Valves Test, Revision 10
- RT-8C, Engineered Safeguards System – Left Channel, Revision 33
- SEP-PLP-IST-102, Inservice Testing of Selected Safety-Related Pumps, Revision 0
- T-218, Service Water Pumps P-7A, P-7B, and P-7C Performance Test by Flow to Containment, Revision 20
- T-302, Emergency Diesel Generator 1-1 Overspeed Trip Setpoint Verification, Revision 10
- WO 52435724, LT-0105 Calibration
- WO 52435755, PCS Hot leg Level LT-0106 Calibration
- WO 52519535, QO-15A, P-52A ISI Test

2RS1 Radiological Hazard Assessment and Exposure Controls

- CR-PLP-2012-02850; Individual Received a Dose Rate Alarm While Moving the Refuel Machine Over the Core; dated April 18, 2012
- CR-PLP-2014-00973; Airborne Radioactivity Excursion in Containment; February 3, 2014
- CR-PLP-2014-00995; Fleet Radiation Procedures Do Not Provide Sufficient Direction; February 3, 2014
- EN-RP-101; Access Control for Radiologically Controlled Areas; Revision 08
- EN-RP-104; Personnel Contamination Events; Revision 6
- EN-RP-105; Radiation Work Permits; Revision 12

- EN-RP-106; Radiological Survey Documentation; Revision 5
- EN-RP-106-01; Radiological Survey Guidelines; Revision 1
- EN-RP-108; Radiation Protection Posting; Revision 13
- EN-RP-115-02; CE SRMP Survey Instructions; Revision 0
- EN-RP-122; Alpha Monitoring; Revision 7
- EN-RP-131; Air Sampling; Revision 11
- EN-RP-208; Whole Body Counting/In-Vitro Bioassay; Revision 5
- Radiation Work Permit and Associated ALARA Files; RWP Number 20140421; 1R23 Insulation Activities
- Radiation Work Permit and Associated ALARA Files; RWP Number 20140429; REFUEL PROJECT: Incore Instrumentation (ICI) Removal/Installation

2RS2 Occupational ALARA Planning and Controls

- ALARA Managers Meeting Minutes; Selected Dates 2012 and 2013
- ALARA Sub-Committee Meeting Minutes; Selected Dates 2012
- EN-OU-100-01; In-Processing Coordinator Duties and Responsibilities; Revision 01
- EN-PL-169; Commitment to ALARA Principles; Revision 00
- EN-RP-100-01; ALARA Initiative Deferrals; Revision 01
- EN-RP-110; ALARA Program; Revision 12
- EN-RP-110-03; Collective Radiation Exposure (CRE) Reduction Guidelines; Revision 02
- EN-RP-110-04; Radiation Protection Risk Assessment Process; Revision 4
- EN-RP-110-05; ALARA Planning and Controls; Revision 2
- Radiation Work Permit and Associated ALARA Files; RWP Number 20120319; Forced Outage Repair of CRD-24 Housing

4OA1 Performance Indicator Verification

- Maintenance Rule Performance Indicator Data, Auxiliary Feedwater, 1st Quarter 2013 through 4th Quarter 2013
- NRC Performance Indicator Technique/Data Sheet, Mitigating System Performance Indicator Heat Removal (AFW), 1st Quarter 2013 through 4th Quarter 2013
- NRC Performance Indicator Technique/Data Sheet, Unplanned Scrams with Complications, 1st Quarter 2013 through 4th Quarter 2013
- Palisades MSPI Basis Document, Section 2.3, Heat Removal System, December 21, 2013

4OA2 Problem Identification and Resolution

- Calculation EA-SP-03316-09, Evaluate Adequacy of Replacing MV-SW135 from Carbon Steel to Stainless Steel and Adding New Flanges, Revision 1
- CEP-FAC-001, Flow Accelerated Corrosion Program Component Scanning and Gridding Standard, Revision 0
- CEP-RR-002, Control and Documentation of ASME Section XI Repair/Replacement Activities, Revision 5
- CH 1.11, Biofouling and MIC Control Program, Revision 4
- CR-PLP-2012-5813, Water Dripping into West ESG from Piping Penetration in Ceiling, August 21, 2012
- CR-PLP-2013-3005, Pinhole Leak Discovered on the Critical Service Water Header that Supplies 1-1 EDG and Control Room HVAC Chiller VC-11, July 10, 2013
- CR-PLP-2013-4613, Pin Hole Leak Discovered on Inlet Side of Valve Body for MV-SW135, E-54A Service Water Outlet CV-0823 Bypass

- CR-PLP-2013-5254, No Inspection of Opportunity Scheduled With Insulation Removal Task for CV-0821, December 12, 2013
- CR-PLP-2013-5278, Potential Multiple Leak Locations on Irrigation Header Isolation Line in Screenhouse, December 14, 2013
- CR-PLP-2014-0270, MV-SW135 Replacement Has Damaged Bevel Ends on Inlet Side of Valve, January 16, 2014
- CR-PLP-2014-0272, Weld Procedure and Process for Replacing MV-SW135 is Incorrect, January 16, 2014
- CR-PLP-2014-0646, Extensive Pitting Identified on Piping Downstream of MV-SW135, January 25, 2014
- CR-PLP-2014-0648, Cavitation Induced Erosion Identified Inside MV-SW135, January 25, 2104
- CR-PLP-2014-0661, Engineering Decided to Remove Additional Piping Due to Extensive Pitting Downstream of MV-SW135, January 25, 2014
- CR-PLP-2014-0773, Elbow on the Extraction Steam Line Going to E-3A Feedwater Heater Did Not Meet the Minimum Calculated Thickness, January 28, 2014
- CR-PLP-2014-1042, Min Wall Violation Identified During FAC Inspection, February 4, 2014
- CR-PLP-2014-1193, Substantial Accumulation of Ooze, Sludge and Bioslimes Found in Non-Critical Service Water Piping During Repair of Leak, February 8, 2014
- CR-PLP-2014-1195, Missed Inspection of Opportunity on Non-Critical Service Water Piping Upstream of MV-SW105, February 8, 2014
- EC No. 39326, Temporary Weld Overlay Repair to HB-23-16" Service Water System Piping (N-661-1 Code Case Repair of Service Water Leak Downstream of CV-0823), Revision 0
- EC No. 47382, Replacement of Six-Inch Blind-Flanged Stub-In Tee Downstream of EDG 1-1 and VC-11 Service Water Supply Isolation Valve, MV-SW377, With a Length of Six-Inch Straight Pipe, Revision 0
- EC No. 48022, Replace Valve MV-SW135 with Stainless Steel Globe Valve, Revision 0
- EC Reply No. 48875, Document Acceptability of Throttling Flow to Component Cooling Water Heat Exchanger E-54B Via Inlet Valve MV-SW134 in Lieu of Outlet Valve CV-0826 to Improve Splashing at CV-0823, Revision 0
- EN-DC-115, Engineering Change Process, Revision 16
- EN-DC-136, Temporary Modification, Revision 10
- EN-DC-315, Flow Accelerated Corrosion Program, Revision 10
- M-208, P&ID Service Water System, Sh. 1A, Revision 62
- M-208, System Diagram for Service Water System, Sheet A, Revision A
- ODMI for Pinhole Leak on MV-SW135, E-54A SW Outlet CV-0823 Bypass, Valve Body, Revision 0
- ODMI for Pinhole Leak on the 6 Inch Flanged Tee on the Critical Service Water Header, Revision 1
- Operability Evaluation for CR-PLP-2013-3005, July 19, 2013
- Operability Evaluation for CR-PLP-2013-4613, October 30, 2013
- Passive Component Review Risk Ranked Issues and Actions
- SEP-SW-PLP-002, Service Water and Fire Protection Inspection Program, Revision 2
- Top Ten Issue Plan for Cavitation in the Service Water System, December 2013
- WI-MSM-M-30, Inspections of Opportunity, Revision 0
- Work Order #324812, CV-0823; Critical Service Water Pipe Leak Downstream of CCW Heat Exchanger, January 23, 2014
- WO 356678, Pin Hole Leak Downstream of MV-SW377 Near Blank Flange, January 31, 2014
- WO 365955, MV-SW135; Pinhole Leak on Inlet Body Weld, January 24, 2014

4OA3 Followup of Events and Notices of Enforcement Discretion

- EN-RP-104, Personnel Contamination Events, Revision 6
- EN-RP-131, Air Sampling, Revision 11
- HP-2.8, Response to Radiological Occurrences, Revision 25
- NUREG 1022, Event Reporting Guidelines, 10 CFR 50.72 and 50.73, Revision 3

LIST OF ACRONYMS USED

ACE	Apparent Cause Evaluation
ADAMS	Agencywide Documents Access Management System
ALARA	As-Low-As-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
BACC	Boric Acid Corrosion Control
CAP	Corrective Action Program
CC	Code Case
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CVC	Chemical and Volume Control
DM	Dissimilar Metal
EDG	Emergency Diesel Generator
EPRI	Electric Power Research Institute
ET	Eddy Current Testing
FOSAR	Foreign Object Search and Retrieval
FME	Foreign Material Exclusion
HPSI	High Pressure Safety Injection
HVAC	Heating Ventilation and Air Conditioning
IEL	Initiating Event Likelihood
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
IST	Inservice Test
LOCA	Loss of Coolant Accident
LOLC	Loss of Level Control
MIC	Microbiologically Induced Corrosion
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
PCP	Primary Coolant Pump
PCS	Primary Coolant System
PI	Performance Indicator
PMT	Post-Maintenance Test
PT	Dye Penetrant Test
PWSCC	Primary Water Stress Corrosion Cracking
RCS	Reactor Coolant System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SG	Steam Generator

SRA	Senior Reactor Analyst
SW	Service Water
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
UT	Ultrasonic Test
VT	Visual Examination
WO	Work Order

cross-cutting aspects, which will be evaluated for cross-cutting themes and potential substantive cross-cutting issues in accordance with IMC 0305 starting with the 2014 mid-cycle assessment review. If you disagree with the cross-cutting aspect assigned, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Palisades Nuclear Plant.

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

Sincerely,

/RA/

Eric Duncan, Chief
Branch 3
Division of Reactor Projects

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Letter to Anthony Vitale from Eric Duncan dated May 7, 2014

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05000255/2014002

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