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

August 8, 2012

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 05000255/2012003

Dear Mr. Vitale:

On June 30, 2012, 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 July 25, 2012, 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, three NRC-identified findings of very low safety significance were identified. The findings involved violations of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

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

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

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

Sincerely,

/RA by Jay A. Lennartz for/

John B. Giessner, Chief
Branch 4
Division of Reactor Projects

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

Enclosure: Inspection Report 05000255/2012003
w/ Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	05000255/2012003
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	April 1, 2012, through June 30, 2012
Inspectors:	T. Taylor, Senior Resident Inspector A. Scarbeary, Resident Inspector J. Ellegood, Senior Resident Inspector, D.C. Cook T. Bilik, Reactor Inspector J. Cassidy, Senior Health Physicist M. Holmberg, Reactor Inspector D. Jones, Reactor Inspector M. Mitchell, Health Physicist S. Shah, Reactor Engineer W. Lyon, Senior Reactor Engineer D. Alley, Senior Materials Engineer
Approved by:	John B. Giessner, Chief Branch 4 Division of Reactor Projects

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000255/2012003; 04/01/2012 – 06/30/2012; Palisades Nuclear Plant; Inservice Inspection Activities; Operability Determinations and Functional Assessments; Refueling and Other Outage Activities

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three Green findings were identified by the inspectors. The findings were considered Non-Cited Violations (NCV) of NRC regulations. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR 50 Appendix B, Criterion III, Design Control, for the failure to operate the Primary Coolant Pumps (PCPs) in accordance with their design operating criteria. In October 2011, a slight rise in vibration levels on the 'C' PCP occurred and was sustained for approximately 24 hours. This was followed by a short spike in vibrations and a return to a lower stabilized value than what had been previously observed. Investigation by the licensee revealed it was likely a piece of an impeller vane which had deformed and broken free. Based on a review of operating experience associated with impellers and further licensee investigation, the inspectors concluded that the PCPs had been operated outside of their license/design basis as stated in the Updated Final Safety Analysis Report (UFSAR) with regard to minimum net positive suction head and maximum flow. Further, based on impeller-like pieces found in the reactor vessel in 2007 (which an apparent cause stated likely came from a PCP), and an operating history which indicated past occurrences of vane breakage and degradation, the inspectors concluded the licensee had the ability to foresee and correct the condition affecting the PCPs prior to the release of a piece in October 2011. The licensee entered the issue in their Corrective Action Program (CAP) as CR-PLP-2011-5744 and performed additional research into the phenomena leading to the impeller degradation. The PCP operating sequence was changed, an Operational Decision Making Issue was implemented, and efforts to explore further procedural changes are on-going to mitigate degradation of the impellers.

  The issue was determined to be more than minor because it impacted the Design Control attribute of the Initiating Events Cornerstone, adversely affecting the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the potential release of impeller pieces in the primary coolant system (PCS) challenges the cornerstone objective. The issue screened as Green, or very low safety significance, based on answering 'no' to the Loss-of-coolant Accident (LOCA) initiator question under the Initiating Events cornerstone in IMC 0609, Attachment 4, Table 4a. This was based

on a review of the licensee's assessment by the regional inspectors, experts at the Office of Nuclear Reactor Regulation (NRR) and Office of Research in determining the deficiency would not likely be an impact to the coolant pressure boundary. The inspectors determined there was no associated cross-cutting aspect because the finding was not indicative of current licensee performance. (1R15)

- Green. The inspectors identified a finding of very low safety significance with an associated NCV of Technical Specification (TS) 5.4.1, Procedures, for the failure to properly follow the work management process for work done to loosen stuck reactor head studs. During the April-May 2012 refueling outage, difficulty was encountered in loosening some of the reactor head studs to support refueling operations. The decision was made to retension the studs that had already been detensioned (without ascending back to Mode 5 from Mode 6) and start over using a more precise electric pumping unit that had not been used to that point due to equipment issues. Contrary to EN-WM-102, Work Implementation and Closeout, the licensee used the field change process, not authorized for this type of change, to "pen-and-ink" different tensioning values and sequence in the normal tensioning procedure (so as not to return to Mode 5). Additionally, the inspectors identified that the steps documented as having been performed as a record of the contingency actions taken differed from what was actually performed. The licensee entered the issue into the CAP as Condition Reports CR-PLP-2012-2610 and CR-PLP-2012-2848, and corrected the contingency work instructions.

  The issue was determined to be more than minor because if left uncorrected, it could lead to more significant safety issues. Specifically, the failure to follow appropriate processes and correctly document reactor head work is indicative of shortfalls that could occur for other safety-related work. Additionally, the licensee was slow to recognize the issue. The inspectors concluded that the Initiating Events Cornerstone was impacted because of the potential for an inadvertent mode change. The finding screened as Green, or very low safety significance, using IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process," based on all of the mitigation criteria being met and no phase 2 or 3 analysis being required per Checklist 3, indicating there was no impact to shutdown safety functions. The inspectors determined that the finding had an associated cross-cutting aspect in the area of human performance in that personnel work practices did not support human performance. Specifically, supervisory and management oversight failed to assure the proper processes were followed (H.4.c). (1R20)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion III, "Design Control" for the licensee's failure to adequately evaluate leaking Safety Injection and Refueling Water Tank (SIRWT) nozzles during the application of American Society of Mechanical Engineers (ASME) Code Case N-705. During the April-May 2012 refueling outage, the SIRWT was drained for inspection and repairs and a deformed nozzle was sealed off, as it was believed to be the potential source of pre-outage leakage. Upon refill, leakage was observed under a different section of the roof upon which the SIRWT rests, indicating a potentially new leak. The licensee employed ASME Code Case N-705 to demonstrate tank operability given the existing leakage and set an upper limit for allowed leakage. Inspector review of the approved evaluation identified certain Code

Case criteria that were not discussed, namely, the residual weld stresses and seismic sloshing stresses. After discussions with the inspectors, the licensee developed residual weld stress values for their evaluation and discussed potential effects of seismic sloshing. The result was a reduction in allowed leakage from 130 gallons per day (gpd) to 34.8 gpd. The licensee entered the issue in their CAP as CR-PLP-2012-04245 and CR-PLP-2012-03732.

The finding was determined to be more than minor because the finding, if left uncorrected, could become a more significant safety concern. The inspectors utilized examples 3j and 3k in IMC 0612, Appendix E, “Examples of Minor Issues,” to inform this determination. Omission of Code-Case-required parameters in the approved evaluation led to reasonable doubt on the operability of the system had the licensee ascended to a mode requiring SIRWT operability. Further analysis was also required by the licensee. Absent NRC identification, the failure to adequately evaluate the leaking SIRWT nozzles could have allowed unstable cracks to remain in service. Unstable nozzle cracks could propagate and allow unacceptable leakage from the SIRWT resulting in loss of inventory and increase the risk for insufficient core cooling for post LOCA conditions. This finding impacted the Mitigating Systems Cornerstone attribute of Equipment Performance (reliability). The finding adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because the licensee promptly corrected this issue and lowered the amount of allowed leakage, the inspectors answered “No” to all of the worksheet questions identified in IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems Cornerstone. The correct leakage limit was in place prior to the required time the tank needed to be operable. Therefore, this finding screened as having very low safety significance (Green). This finding has a cross-cutting aspect in the area of Human Performance for the work practices component. The licensee did not provide adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported (H.4.c). Specifically, the licensee failed to ensure that the vendor evaluation to demonstrate SIRWT nozzle integrity with through-wall cracks included consideration of residual weld stresses and seismic sloshing stresses. The inspectors determined the primary cause of this finding based upon discussions with the licensee’s engineering staff. (1R08)

B. <u>Licensee-Identified Violations</u>

No violations were identified.

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period operating at or near 100 percent power until April 8, 2012, when the licensee commenced a shutdown of the plant for a planned refueling outage (1R22). The plant restarted and synchronized back to the grid on May 12, 2012. The plant then operated at full power until it shut down on June 12, 2012 to address leakage from the safety injection and refueling water tank (SIRWT). The plant remained shut down through the end of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors' review included:

- The coordination between the TSO and the plant during off-normal or emergency events;
- The explanations for the events;
- The estimates of when the offsite power system would be returned to a normal state; and
- The notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- The actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- The compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;
- A re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and

- The communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

Documents reviewed are listed in the Attachment to this report. The inspectors also reviewed Corrective Action Program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures.

This inspection constituted one readiness of offsite and alternate AC power systems sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- ‘A’ service water system with ‘B’ service water out for maintenance; and
- ‘B’ and ‘C’ instrument air systems with ‘A’ instrument air out of service for maintenance.

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

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

b. Findings

No findings were identified.

.2 Semiannual Complete System Walkdown

a. Inspection Scope

On June 22, 2012, the inspectors performed a complete system alignment inspection of the Emergency Diesel Generator System to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

These activities constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings were identified.

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 availability, accessibility, and the condition of firefighting equipment in the following risk-significant plant areas:

- Fire Area 9: screenhouse/intake structure;
- Fire Areas 2 & 3: cable spreading room and 1-D switchgear room;
- Fire Area 23: turbine building/elev. 607', 612', 625';
- post-indicating valves on the outside fire protection header; and
- transformer deluge.

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee's fire plan.

The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant's Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant's ability to respond to a security event. Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee's CAP. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08P)

From April 17, 2012 through April 27, 2012 the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, steam generator (SG) tubes, emergency feedwater systems, risk-significant piping and components and containment systems.

Additionally, from May 6 through May 8, 2012, the inspectors conducted a review of the licensee's response to degradation of the risk significant piping and components associated with the American Society of Mechanical Engineers (ASME) Code Class 2 safety SIRWT. This included the licensee's implementation of ASME Code Case N-705.

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 reviewed the following non-destructive examinations mandated by the 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 if these were dispositioned in accordance with the ASME Code or an NRC approved alternative requirement.

- ultrasonic examination of the engineering safeguards system elbow to pipe weld ESS-6-SIS-2A1-5, Report Number 1R22-UT-12-007;
- visual examination of the reactor vessel closure head, Report Number 1R22-VT-12-157;
- liquid penetrant examination of the engineering safeguards system, pipe to tee weld ESS-2-SIS-1A1-4; Report Number 1R22-PT-12-007; and
- visual examination of the primary coolant system nozzle to pipe PCS-014 Weld 6 (PC1024), Report Number 1R22-VT-12-036.

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

- evaluation (CR-PLP-2011-02413) of relevant indications found during the visual examination (Report Number VT-10-088) of the reactor head; and
- evaluation (CR-PLP-2012-03344) of relevant indications found during the visual examination (Report Number 1R22-VT-12-157) of the reactor head.

The licensee had not performed pressure boundary welding since the beginning of the preceding outage. Therefore, no NRC review was completed for this inspection procedure attribute.

On April 29, 2012 the licensee identified a 2.7 to 4.6 gallons per day (gpd) leak from the SIRWT (Section XI Code Class 2) collected from below the three inch diameter "G" and "H" nozzles. Because these nozzles are located at the tank floor the licensee would need to drain the tank to effect repairs. Instead, the licensee elected to apply Code Case N-705 "Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks" and defer repair of this flaw until the next refueling outage. The inspectors reviewed records supporting the licensee's application of Code Case N-705 to determine if the licensee had demonstrated structural integrity of the leaking SIRWT nozzles.

b. <u>Findings</u>

<u>Inadequate Design Margins for Evaluation of Leaking Safety Injection and Refueling Water Tank Nozzles</u>

<u>Introduction</u>: A finding of very low safety significance (Green) and associated Non-Cited Violation (NCV) of 10 CFR 50 Appendix B, Criterion III, Design Control was identified by the inspectors for the licensee's failure to adequately evaluate the leaking SIRWT nozzles. Specifically, the licensee failed to specify or discuss weld residual stress and seismic sloshing loads in the evaluation to demonstrate stability of a through-wall crack in the limiting area of the SIRWT in accordance with ASME Code Case N-705.

<u>Description</u>: On May 6, 2012, the inspectors identified that the licensee had not included residual weld stress values nor consideration of seismic sloshing stresses in the evaluation intended to demonstrate that a presumed crack causing leakage in the limiting nozzles of the SIRWT would remain stable for continued service. The inspectors were concerned that without adequate demonstration of crack stability margins, the cracked nozzles could fail in-service and affect the SIRWT function to provide the cooling water supply for post-LOCA core cooling.

The licensee identified approximately 2 to 6 gpd leakage from the bottom of the SIRWT following the refueling outage. The actual source of this leakage was from the floor area of the tank, which was inaccessible, but based upon previous history with nozzle leakage, the licensee presumed that the 'G' and/or 'H' nozzle could be cracked and leaking. The licensee elected to apply an NRC approved Code Case N-705, "Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks," and deferred repair of this flaw until the next refueling outage. The method selected by the licensee under Code Case N-705 was the Bounding Flaw

Evaluation described in Section 2.4 of this Code Case. To support this method, a critical stress intensity value in Mode 1 loading (K_{1C}) must be compared to the K calculated for a limiting crack, in accordance with Section 3.1 of the Code Case. Sections 3.1(b) and (e) required that the stress intensity include residual stresses resulting from original welding. Section 3.1(a) required seismic sloshing stresses. The stress intensity factor (K) is used in fracture mechanics to more accurately predict the stress state (e.g., stress intensity) near the tip of a crack caused by a remote load or residual stresses. To apply Code Case N-705 the licensee approved a vendor evaluation RPT-12-0097, "Evaluation of SIRW Tank T-58," revision 0, which relied on a supporting vendor calculation 1100772.301, "Fracture Mechanics Evaluation of Hypothetical Cracks in SIRW Tank Bottom Nozzles." These evaluations were used to determine a bounding allowable circumferential crack size of 1.12 inches in the 'G' nozzle, which the licensee determined would leak at 130 gpd.

The inspectors identified that the licensee had accepted a vendor evaluation that did not include a value, nor discuss, residual weld and seismic sloshing stresses in the determination of stress intensity for the limiting crack. This resulted in a non-conservative calculation of the allowable crack size. With this larger non-conservative crack size, the licensee evaluation accepted higher levels of in-service leakage from the SIRWT. If the crack(s) became unstable before reaching this non-conservative maximum leakage rate, they could propagate around the nozzle weld, potentially creating a 3 inch diameter non-isolable hole in the bottom of the SIRWT. Without consideration of these stresses, the licensee evaluation had not demonstrated adequate structural design margins for the leaking SIRWT nozzles. In response, the licensee entered this issue into the CAP as CR-PLP-2012-04245 and CR-PLP-2012-03732 and recalculated the (K_{1C}) and allowable flaw size. This reduced the allowable leakrates in their Operational Decision Making Issue associated with CR-PLP-2012-03368. Specifically, the bounding allowable flaw size was changed from 1.12 inches to 0.33 inches for the 'G' nozzle. This changed the maximum allowable leakage from 130 gpd to 34.8 gpd. Seismic sloshing was determined to have a negligible effect when analyzed. Later, the licensee performed more in-depth analyses of stresses at the SIRWT nozzles, which resulted in an allowed flaw size still lower than had originally been calculated (0.73 inches).

On June 12, 2012, the licensee shutdown the plant when an administrative limit of 31 gpd of SIRWT leakage was identified. Inspection and repair activities lasted approximately 1 month. During maintenance activities in the tank with the plant shutdown, improper placement of catches during use of a water-cooled tool resulted in some water leakage into the control room. Inspectors were on site to assess the issue and observed the corrective actions taken to address the leakage pathway. Any performance deficiencies associated with improper maintenance will be followed up by the inspectors in the third quarter.

Analysis: The inspectors determined that the licensee's failure to adequately evaluate the leaking SIRWT nozzles in accordance with ASME Code Case N-705 was a performance deficiency. The finding was determined to be more than minor because the finding, if left uncorrected, could become a more significant safety concern. The inspectors utilized examples 3j and 3k in IMC 0612, Appendix E, "Examples of Minor Issues," to inform this determination. Omission of Code-Case-required parameters in the approved evaluation led to reasonable doubt on the operability of the system had the licensee ascended to a mode requiring SIRWT operability. Further analysis was also

required by the licensee. Absent NRC identification, the failure to adequately evaluate the leaking SIRWT nozzles could have allowed unstable cracks to remain in service. Unstable nozzle cracks could propagate and create a 3 inch diameter hole at the bottom of the SIRWT resulting in loss of inventory and increase the risk for insufficient core cooling for post-LOCA conditions. This finding impacted the Mitigating Systems Cornerstone attribute of Equipment Performance (reliability). The finding adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Because the licensee promptly corrected this issue before potentially unstable cracks were returned to service, the inspectors answered “No” to all of the worksheet questions identified in IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table 4a for the Mitigating Systems Cornerstone. The correct leakage limit was in place prior to the required time the tank needed to be operable. Therefore, this finding screened as having very low safety significance (Green).

This finding has a cross-cutting aspect in the area of Human Performance for the work practices component. The licensee did not provide adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported (H.4.c). Specifically, the licensee failed to ensure that the vendor evaluation to demonstrate SIRWT nozzle integrity with through-wall cracks included consideration of residual weld stress and seismic sloshing stresses. The inspectors determined the primary cause of this finding based upon discussions with the licensee’s engineering staff.

<u>Enforcement</u>: Title 10 CFR 50, Appendix B, Criterion III, ”Design Control,” requires, in part, that design control measures provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. An alternative calculational method described in Section 3.1.b of ASME Code Case N-705 required, in part, that “residual stresses resulting from original welding and any rework, bolt-up stresses, and cladding-induced stresses shall be evaluated in accordance with the methods of A-3200.” ASME Section XI, Appendix A, Paragraph A-3200 required, in part, that “residual stresses and applied stresses from all forms of loading, including pressure stresses and cladding-induced stresses, shall be considered.

Contrary to the above, on May 6, 2012, the licensee’s design control measures failed to verify the adequacy of the design of nozzles postulated to be degraded in the SIRWT, in that the methodology and design inputs used to implement ASME Code Case N-705 did not consider a significant factor (e.g., residual weld stress) which affected the structural integrity of the nozzles. Specifically, the licensee approved PLP-RPT-12-0097, “Evaluation of SIRW Tank T-58,” Revision 0, which relied on a supporting vendor calculation 1100772.301, “Fracture Mechanics Evaluation of Hypothetical Cracks in SIRW Tank Bottom Nozzles,” which did not include values for, nor discussion of, weld residual stress. Since the licensee revised the evaluation to include consideration of weld residual stress, and the less significant seismic sloshing stress, an immediate safety hazard no longer existed. Because this violation was of very low safety significance and it was entered into the licensee’s corrective action program as CR-PLP-2012-04245 and CR-PLP-2012-03732, it is being treated as a NCV, consistent

with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2012003-01, Inadequate Design Margins for Evaluation of Leaking SIRWT Nozzles).

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities

a. Inspection Scope

For the reactor vessel head, a bare metal visual examination and a non-visual examination 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 54 penetration nozzles to determine if the activities were conducted in accordance with the requirements of ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, to determine:

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

The inspectors observed and reviewed data for the non-visual examinations (Ultrasonic and Eddy Current) conducted on the reactor vessel head penetrations to determine if the activities were conducted in accordance with the requirements of ASME Code Case N-729-1 and 10 CFR 50.55a(g)(6)(ii)(D). Specifically, to determine:

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

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

b. Findings

No findings were identified.

.3 Boric Acid Corrosion Control

a. Inspection Scope

The inspectors performed an independent walkdown of all portions of accessible containment systems which had received a recent licensee boric acid walkdown and verified whether the licensee's Boric Acid Corrosion Control visual examinations emphasized locations where boric acid leaks can cause degradation of safety significant components.

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

- CR-PLP-2012-02284 with Evaluation No. 12-PAL-0062, Boric Acid Discovered on MO-3062 HPSI TRN 2 Loop 2B packing area;
- CR-PLP-2012-02286 with Evaluation No. 12-PA-0064, Boric Acid Discovered on MV-ES3047 T-82C Pressure Control CV-3047 Inlet; and
- CR-PLP-2012-02290 with Evaluation No. 12-PAL-0068, Boric Acid discovered on CRD-20 Control Rod Drive Mechanism.

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

- CR-PLP-2012-02294, Boric Acid Discovered on MO-3049 Safety Injection Tank T-82C Outlet Isolation, Work Request 268867; and
- CR-PLP-2012-02589, Boric Acid Leak on Backside of Fitting FT-0309, Low Pressure Safety Injection Flow Loop 1B, Work Request 269503.

b. Findings

No findings were identified.

.4 Steam Generator Tube Inspection Activities

a. Inspection Scope

The NRC inspectors observed acquisition of eddy current (ET) data, interviewed ET data analysts, and reviewed documentation related to the SG ISI program to determine if:

- In-situ SG tube pressure testing screening criteria used were consistent with those identified in the Electric Power Research Institute (EPRI) TR-107620, Steam Generator In-Situ Pressure Test Guidelines and that these criteria were properly applied to screen degraded SG tubes for in-situ pressure testing;
- in-situ pressure test records demonstrated pressure and hold times consistent with EPRI TR-107620, In-situ Pressure Test Guidelines;
- in-situ pressure test results were properly applied to SG tube integrity performance criteria identified in EPRI TR-107621;

- the numbers and sizes of SG tube flaws/degradation identified was bound by the licensee's previous outage Operational Assessment predictions;
- the SG tube ET examination scope and expansion criteria were sufficient to meet the TSs, and the EPRI 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 6;
- the SG tube ET examination scope included potential areas of tube degradation identified in prior outage SG tube inspections and/or as identified in NRC generic industry operating experience applicable to these SG tubes;
- the licensee identified new tube degradation mechanisms and implemented adequate extent of condition inspection scope and repairs for the new tube degradation mechanism;
- the licensee implemented repair methods which were consistent with the repair processes allowed in the plant TS requirements and to determine if qualified depth sizing methods were applied to degraded tubes accepted for continued service;
- the licensee implemented an inappropriate "plug on detection" tube repair threshold (e.g., no attempt at sizing of flaws to confirm tube integrity);
- the licensee primary-to-secondary leakage (e.g., SG tube leakage) was below 3 gpd 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 1003138, Pressurized Water Reactor Steam Generator Examination Guidelines, Revision 6; and
- the licensee performed secondary side SG inspections for location and removal of foreign materials.

b. Findings

No findings were identified.

.5 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI/SG related problems entered into the licensee's corrective action program and conducted interviews with licensee staff to determine if:

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

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

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On May 29, 2012, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training to verify that 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;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

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

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

b. Findings

No findings were identified.

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

On May 3, 2012, the inspectors observed operations staff conducting activities in the control room during the refueling outage to drain the primary coolant system to reduced inventory and then subsequently fill the PCS during a vacuum fill evolution. This was an infrequently performed task or evolution that required heightened awareness and was related to an increase in risk. The inspectors evaluated the following areas:

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

- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications (if applicable).

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

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

a. 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:

- 4160V electrical system;

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

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

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

This inspection constituted one quarterly maintenance effectiveness sample 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:

- steam generator nozzle dam leakage;
- response to bent incore stalk;
- SIRWT repairs during forced outage; and
- emergent procedure changes to emergency core cooling system check valve testing.

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

Specific documents reviewed during this inspection are listed in the Attachment to this report. These maintenance risk assessments and emergent work control activities constituted four samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- ‘C’ Primary Coolant Pump (PCP) impeller issues;
- repairs to safety injection/refueling water tank; and
- primary-side steam generator manway installations.

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and UFSAR to the licensee’s evaluations to determine whether the components or systems were operable. Where compensatory measures

were required to maintain operability, the inspectors determined whether the measures in place would function as intended and were properly controlled. The inspectors determined, where appropriate, compliance with bounding limitations associated with the evaluations. Additionally, the inspectors reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

Introduction: A finding of very-low safety significance (Green) and associated NCV of 10 CFR 50 Appendix B, Criterion III, Design Control, was identified by the inspectors for the failure to operate the PCPs in accordance with their design operating criteria. Specifically, the PCPs were operated under conditions that allowed degradation of the leading edges of the pump impellers to occur, which resulted in fragments being released into the PCS.

Description: On October 11, 2011, operators noted a step change of approximately 1 mil in the ‘C’ PCP vibration monitor. Vibration values were trending around 10.5 mils prior to the perturbation. Vibration indication remained steady at the new level for approximately 25 hours, at which point a vibration spike to 22 mils occurred for less than three seconds. The vibration level then abruptly returned to a value slightly below the normal level, and remained there for the rest of the operating cycle. Slight changes in flow, coolant temperature, and pump current were seen concurrent with the initial vibration change and remained within acceptable values. Investigation by the licensee with the assistance of outside consultants concluded it was likely that a piece of the ‘C’ PCP impeller deformed and broke free. There was no indication of degradation to the primary coolant system or reactor core components as a result of this postulated failure. NRC inspectors, including experts at the Offices of Research and Nuclear Reactor Regulation (NRR) reviewed the data gathered by the licensee and concluded that the pump was safe to operate until the refueling outage in April 2012 with the monitoring plan that the licensee had put in place.

In further consultation with industry specialists over the next several months, the licensee reviewed previous site and industry operating experience regarding PCP impeller issues and assessed the manner in which the Palisades pumps were operated. The licensee identified impeller cracking had been observed at Palisades on several occasions since 1984, when the pumps had been removed for inspection and refurbishment/replacement. Additionally, pieces suspected to be from impellers were discovered in the bottom of the reactor vessel in 1984 and 2007. The research concluded that the cause of the failures is fatigue-related effects from the operation of the pumps in conditions beyond the maximum flow rates and below the minimum net positive suction head recommendations as described in the UFSAR and other design documentation. These conditions are present when operating only one or two PCPs during reduced temperatures and pressures (typically 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 vane cracking and the release of impeller fragments. The licensee noted, based on metallurgical examination of a previous fragment, previous pump inspection findings,

and the mechanism by which the cracks propagate, that weld-refurbished impellers were particularly susceptible to degrading to a point where a piece could be released. Currently, none of the PCPs contain any remaining weld-repaired impeller areas (ones that did are postulated to have released pieces already). Also, at normal operating temperature and pressure, there is adequate net positive suction head on all PCPs, so these additional stresses are not present.

In response to the discovery of two pieces that resembled the PCP impeller composition during reactor vessel inspections in 2007, the licensee conducted an apparent cause analysis. The conclusion was that the pieces were most likely from the ‘D’ PCP. Additionally, the analysis explored the history of Palisades’ PCP impeller conditions which included repeat occurrences of cracking having been identified and an instance of “heavy recirculation damage,” which rendered an impeller unfit for continued use. The pump manufacturer, Flowserve, also released a Tech Alert due to the Palisades PCP vane cracking history. The apparent cause analysis implied that the pieces were fatigue generated and that additional vane breakage was possible. Despite this, the PCPs were not declared as non-conforming nor were any compensatory measures taken. When the ‘D’ PCP was later inspected after removal during the 2009 refueling outage, it did not have any pieces of impeller missing. Inspections of the other PCPs, which were recommended in the apparent cause and had been planned to be executed if the ‘D’ PCP was not the source of the 2007 pieces, were cancelled. The cancellations were based, in part, on thoughts that the pieces may have originated elsewhere. However, vessel inspections done in 2007 revealed no deficiencies that would infer the pieces were generated somewhere within the reactor vessel, and the 2007 apparent cause analysis had essentially ruled out other sources.

In response to the October 2011 event and subsequent research conducted to better understand the phenomena affecting the PCPs, the licensee has instituted a monitoring plan, changed the preferred sequence for starting/stopping PCPs during startups and shutdowns, and has corrective actions to explore further procedure changes regarding operation of the PCPs and the resultant impact on other aspects of plant operation.

Since the licensee was intending to have this non-conformance on the C pump (missing impeller pieces) the entire cycle, the inspectors (including experts at the Offices of Research and NRR) reviewed the impact of this non conformance on the PCP safety functions. Key safety functions of the pump are to provide a coolant pressure boundary and ensure an adequate coast down of flow. The review indicated there were no current safety issues with this non-conformance. The inspectors are evaluating the monitoring plan to determine its long-term effectiveness.

Analysis: The inspectors determined that operation of the PCPs in a manner known to cause impeller degradation to the point of potential fragment release without corresponding monitoring or controls, as would be required by 10 CFR 50 Appendix B, Criterion III, Design Control, was a performance deficiency warranting further evaluation in the SDP. The issue was determined to be more than minor because it impacted the Design Control attribute of the Initiating Events Cornerstone, adversely affecting the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, potential release of impeller pieces in the PCS challenges the cornerstone objective. The issue screened as Green, or very low safety significance, based on answering ‘no’ to the LOCA initiator question under the Initiating Events Cornerstone in

IMC 0609, Attachment 4, Table 4a. This was based on a review of the licensee's assessment by the regional inspectors, experts at NRR and Office of Research in determining the deficiency would not likely be an impact to the coolant pressure boundary. The inspectors determined there was no associated cross-cutting aspect because the finding was not indicative of current licensee performance.

Enforcement: 10 CFR 50, Appendix B, Criterion III requires, in part, that measures shall be established to assure that regulatory requirements and the design bases, for SSCs subject to Appendix B, are correctly translated into specifications, drawings, procedures, and instructions. It states further that deviations from such standards shall be controlled. Contrary to this requirement, the licensee operated the PCPs (safety-related components) in a manner outside the design basis without requisite controls. Specifically, PCPs known to have impellers prone to degradation that could release fragments did not have appropriate assessments or controls performed to address this deviation from design requirements. The inspectors concluded the licensee was not in compliance with Criterion III since at least as early as 2007, the date the when metallic pieces were found in the reactor vessel. It has been a performance deficiency since June 16, 2008, when the last corrective action was closed documenting metallic pieces were found in the reactor vessel. Given the substantial operating experience that had been reviewed concerning the impellers, the inspectors concluded it was within the licensee's ability to foresee and correct that PCP operation deviated from the established design basis. In October 2011, in response to the suspected release of another impeller piece, the licensee instituted a monitoring plan for the PCPs and explored different sequencing of PCP operation during subsequent startups and shutdowns (which were implemented in the future outages in an effort to mitigate the identified design deficiencies with the PCPs). The licensee also entered the issue in the CAP as CR-PLP-2011-05744. Because this violation was of very low safety significance and it was entered into the licensee's CAP it is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2012003-02, Operation of Primary Coolant Pumps Outside Design Basis).

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following modifications:

- temporary service water modification for discharge pipe replacement;
- modifications to the SIRWT; and
- battery breaker replacements.

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

modification with operations, engineering, and training personnel to ensure that the individuals were aware of how the operation with the plant modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted two temporary modification samples as defined in IP 71111.18-05. Inspection of the modifications to the SIRWT continued into the third quarter of 2012, therefore, those activities do not count as a sample this quarter and will be counted next quarter.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- ED-06, Inverter #1 preventive maintenance and overhaul;
- replacement of reactor protection system contactors;
- main steam isolation valve inspections and repairs;
- pressurizer spray valves refurbishments;
- replacement of containment high pressure switches;
- control rod drive position verification after control rod drive repairs;
- use of temporary tanks for SIRWT draining; and
- SIRWT repairs.

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

This inspection constituted eight post-maintenance testing samples as defined in IP 71111.19-05. Inspection of the SIRWT repairs continued into the third quarter of 2012, therefore, those activities do not count as a sample this quarter and will be counted next quarter.

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Refueling Outage Activities

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the 1R22 refueling outage (RFO), conducted from April 8 to May 12, 2012, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out of service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities.
- Monitoring of decay heat removal processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Licensee fatigue management, as required by 10 CFR 26, Subpart I.
- Refueling activities, including fuel handling and sipping to detect fuel assembly leakage.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the primary containment to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to RFO activities.

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

b. Findings

Introduction: A finding of very low safety significance (Green) with an associated NCV of TS 5.4.1, Procedures, was identified by the inspectors for the failure to properly follow the work management process for work done to loosen stuck reactor head studs. Specifically, the field change process was employed and the documented work did not align with what was actually performed in the field.

Description: On April 15, 2012, during the 1R22 Refueling Outage, the inspectors learned that during reactor head detensioning activities to transition to Mode 6 the previous night, workers encountered difficulty in trying to loosen some of the studs. An issue had developed earlier with the preferred electric hydraulic pumping unit which is used to provide the necessary force to detension studs. As a result, an air-operated pump (allowed by procedure) was utilized for the detensioning process. During the process, one of the studs could not be detensioned. Procedurally allowed contingencies were taken to release the stud successfully, but the next studs in the sequence were also difficult to detension. At this point, the licensee stopped work to determine what the issue was and how to proceed. The decision was made to perform a partial retensioning of the head such that the plant would stay in Mode 6 (i.e., not all studs fully tensioned), and then begin detensioning again using the electric hydraulic pumping unit (which had been repaired from the previous issue and was believed to be a more precise tool to use).

The inspectors reviewed the procedures to perform this contingency action and found that the work order had no work instructions in it. The inspectors were told this was a contingency work order and that the steps that were performed would be entered after the contingency was complete or done in parallel. The inspector asked what the workers were using to perform the task and was told that pen-and-ink changes were made to the governing procedure. While procedure EN-WM-102, "Work Implementation and Closeout," allows field changes to be performed, they are only allowed when work scope does not change (i.e., detail clarifications or enhancements). The inspector disagreed that a field change was acceptable based on the potential to re-enter Mode 5. Additionally, although the normal retensioning procedure was utilized, the sequencing and tension values to address the current situation were different since the licensee was not starting from a fully detensioned condition and they were not going to fully retension the head. The inspector observed the activity (already in progress) and concluded the contingency plan was followed as intended and there was no inadvertent transition back to Mode 5. However, the inspector reviewed the contingency work instruction that was created to document the work performed and found numerous errors between it and the steps actually performed in the field. In later discussions, the licensee indicated the proper processes were not followed and outlined several alternatives that could have been used to properly review and document their desired course of action. Later, the inspector reviewed the condition report written to address the issue and found it did not address the non-compliance with the work management process. Subsequently, the licensee added a corrective action to the previous condition report to address this issue.

Analysis: The failure to follow the work management process for activities associated with emergent work on the reactor head was a performance deficiency which warranted further evaluation using the SDP. The issue was determined to be more than minor because if left uncorrected, it could lead to more significant safety issues. Specifically, the failure to follow appropriate processes to plan and document reactor head work

could occur for other safety-related work. Additionally, the licensee was slow to recognize the issue. The inspectors concluded that the Initiating Events cornerstone was impacted because of the potential for an inadvertent mode change. The finding screened as Green, or very low safety significance, using IMC 0609, Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process," based on all of the mitigation criteria being met and no phase 2 or 3 analysis being required per Checklist 3 as no shutdown safety functions were impacted.

The inspectors determined that the finding had an associated cross-cutting aspect in the area of human performance in that personnel work practices did not support human performance. Specifically, supervisory and management oversight failed to assure the proper processes were followed (H.4.c). Additionally, the station was not timely in recognizing and addressing the issue of concern.

Enforcement: TS 5.4.1, Procedures, states that written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33 procedures include those associated with maintenance on safety-related equipment and refueling. EN-WM-102, "Work Implementation and Closeout," is the site procedure for planning and documenting safety related maintenance. Contrary to TS 5.4.1, on April 15, 2012, the licensee failed to properly plan contingency work in accordance with EN-WM-102 on the reactor head to loosen stuck studs. Specifically, a field revision was utilized to mark-up a procedure when more rigorous review and approvals should have been employed. The licensee generated CR-PLP-2012-02848 to address the issue. Because this violation was of very low safety significance and it was entered into the licensee's CAP, it is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2012003-03, Failure to Follow Work Management Process for Reactor Head Work)

.2 Other Outage Activities: Forced Outage due to Safety Injection and Refueling Water Tank Leakage

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on June 12, 2012, and continued through the end of the quarter. The licensee shut the plant down when leakage from the SIRWT exceeded the administrative limit of 31 gpd. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, control and monitoring of decay heat removal, and control of outage activities.

Since the outage extended into the third quarter of 2012, these activities do not count as an inspection sample this quarter but will be counted next quarter.

b. Findings

No findings were identified.

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:

- QO-14B, ‘B’ SW pump inservice test (inservice test);
- RT-8C/D, engineered safeguards system – left/right channel (routine);
- RO-216, service water flow verification (routine);
- QO-6/RO-112, reactor head/pressurizer vent valves surveillance testing (routine);
- RO-105, safety injection tank check valves full flow testing (routine);
- RO-65, high pressure safety injection check valve and flow balance testing (routine); and
- RO-32-65, LLRT – Local Leak Rate Test Procedure for Penetration MZ-65 (containment isolation valve).

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, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the USAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, ASMEs code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;

- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational 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 radiological surveys from selected plant areas and evaluated whether the thoroughness and frequency of the surveys where appropriate for the given radiological hazard.

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

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

- Repack Pressurizer Spray Controls CV-1057 and CV-1059;
- Refuel Project: Rx and Spent Fuel Pool Tilt Pit Activities;
- Refuel Project: Rx Cavity Decon and Support Activities;
- Refuel Project: Rx Vessel Disassembly; and
- S/G Primary Side Activities.

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;
- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

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

b. Findings

No findings were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

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

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

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

b. Findings

No findings were identified.

.4 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, radiation work permits, and worker briefings.

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

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

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

The inspectors reviewed the following radiation work permits for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 20120468, Repack Pressurizer Spray Controls CV-1057 and CV-1059;
- RWP 20120431; Refuel Project: Rx and Spent Fuel Pool Tilt Pit Activities;
- RWP 20120432; Refuel Project: Rx Cavity Decon and Support Activities;
- RWP 20120433; Refuel Project: Rx Vessel Disassembly; and
- RWP 20120454; S/G Primary Side Activities.

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

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

b. Findings

No findings were identified.

.5 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 radiation work permit controls/limits in place, and whether their performance reflected the level of radiological hazards present.

b. Findings

No findings were identified.

.6 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors observed the performance of the radiation protection technicians with respect to all radiation protection work requirements. The inspectors evaluated whether technicians were aware of the radiological conditions in their workplace and the radiation work permit 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-Is-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 man-hour estimates) for the current annual collective exposure estimate for reasonable accuracy for select as-low-as-is-reasonably-achievable (ALARA) work packages. The inspectors reviewed applicable procedures to determine the methodology for estimating exposures from specific work activities and the intended dose outcome.

The inspectors evaluated whether the licensee had established measures to track, trend, and if necessary, to reduce occupational doses for ongoing work activities. The inspectors assessed whether trigger points or criteria were established to prompt additional reviews and/or additional ALARA planning and controls.

b. Findings

No findings were identified.

.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, or high radiation areas. The inspectors evaluated whether workers demonstrated the ALARA philosophy in practice (e.g., workers are familiar with the work activity scope and tools to be used, workers used ALARA low-dose waiting areas) and whether there were any procedure compliance issues (e.g., workers are not complying with work activity controls). The inspectors observed radiation worker performance to assess whether the training and skill level was sufficient with respect to the radiological hazards and the work involved.

b. Findings

No findings were identified.

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

This inspection constituted a partial sample as defined in IP 71124.03-05.

.1 Engineering Controls (02.02)

a. Inspection Scope

The inspectors reviewed airborne monitoring protocols by selecting installed systems used to monitor and warn of changing airborne concentrations in the plant and evaluated whether the alarms and set-points were sufficient to prompt licensee/worker action to ensure that doses are maintained within the limits of 10 CFR Part 20 and the ALARA concept.

The inspectors assessed whether the licensee had established trigger points (e.g., the EPRI's, "Alpha Monitoring Guidelines for Operating Nuclear Power Stations") for evaluating levels of airborne beta-emitting (e.g., plutonium-241) and alpha-emitting radionuclides.

b. Findings

No findings were identified.

.2 Use of Respiratory Protection Devices (02.03)

a. Inspection Scope

The inspectors selected several individuals assigned to wear a respiratory protection device and observed them donning, doffing, and functionally checking the device as appropriate. Through interviews with these individuals, the inspectors evaluated whether they knew how to safely use the device and how to properly respond to any devices malfunction or unusual occurrence (loss of power, loss of air, etc.).

b. Findings

No findings were identified.

3. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, and Emergency Preparedness

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours performance indicator To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in the Nuclear Energy Institute Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's operator narrative logs, issue reports, event reports and NRC Inspection Reports for the period of second quarter 2011 through the first quarter of 2012 to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned scrams per 7000 critical hours 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 that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: identification of the problem was complete and accurate; timeliness was commensurate with the safety significance; evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes,

extent-of-condition reviews, and previous occurrences reviews were proper and adequate; and that the classification, prioritization, focus, and timeliness of corrective actions were commensurate with safety and sufficient to prevent recurrence of the issue. Minor issues entered into the licensee's CAP as a result of the inspectors' observations are included in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Selected Issue Followup Inspection: Substantive Cross-Cutting Issue in Oversight (H.4.c)

a. Inspection Scope

During the 2011 end-of-cycle review, the NRC noted that licensee performance had resulted in three cross-cutting themes. One of the themes was associated with Oversight under the Work Practices component of the Human Performance cross-cutting area (H.4.c). The inspectors reviewed the causal evaluations and status of corrective actions dealing with this theme.

For Oversight, the inspectors reviewed the Root Cause Evaluation, Common Cause Evaluation, training materials, corrective action documentation, recovery plan, and NRC inspection findings. The licensee determined that the root cause of management oversight contributing to plant issues was that managers and supervisors have focused more on work practices rather than on supervisory oversight and employee development. The contributing cause identified was that some managers and supervisors do not always understand that what they perceive as distractions are part of their roles and responsibilities. The inspectors concluded that based on the findings reviewed by the licensee, interview results, condition report trend code searches, and

events tagged with management oversight components, that the licensee identified reasonable root and contributing causes. Corrective Actions included training for the Leadership Development Program on expectations for managers (and supervisors during outages), the use of "What It Looks Like," or WILL-sheets for management observations and an expectation for the number of management observations conducted monthly, a re-delegation of "lower level" duties, filling of managerial/supervisory vacancies, and re-instituting monthly Leadership and Alignment meetings. In reviewing licensee performance over the first and second quarters of 2012, the inspectors noted that issues are still being encountered in the area of management oversight. Multiple condition reports, observations by the inspectors, and causal evaluations conducted on issues that arose during the most recent refueling outage had identified insufficient oversight and vendor control as a contributor to re-work activities that increased the accumulated dose for the outage. An example of this was during the first attempt to replace the reactor vessel head onto the flange by a set of new contract workers who were not familiar with the intricacies of the Palisades-specific configuration. The result was that an Incore Instrument stalk did not enter the head nozzle correctly, bending this stalk, and creating a situation that led to re-conducting the evolution and accumulating significantly more dose to the workers who had to fix the stalk as well as re-conduct the activity. The only Palisades oversight person was watching the evolution from a far distance and could not see into the reactor cavity where the activity was taking place. This person was also not involved in the briefings prior to the evolution. Additional weaknesses in oversight are described in this report as findings in Sections 1R08 and 1R20. The inspectors also noted the site's Quality Assurance organization rated vendor oversight as a "Top 3" issue and cited examples similar to inspector observations during the refueling outage. The aforementioned issues, along with other low-level events, illustrate a continued need for improvement in the area of oversight. This substantive cross-cutting issue will be further reviewed as part of the NRC's mid-cycle assessment process.

This review constituted a single selected issue for followup inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.4 Selected Issue Followup Inspection: Substantive Cross-Cutting Issue in Conservative Assumptions (H.1.b)

a. Inspection Scope

During the 2011 end-of-cycle review, the NRC noted that licensee performance had resulted in three cross-cutting aspect themes. One of the themes was in the aspect dealing with Conservative Assumptions under the Decision Making component of the Human Performance cross-cutting area (H.1.b). The inspectors reviewed the causal evaluations and status of corrective actions dealing with this theme.

The inspectors discovered during their review that some of the corrective actions developed by the licensee had either been incorrectly stated as complete or were still pending completion. One of the actions was to develop and implement a technical human performance tool for use by both individuals and observers for validating effective

use of conservative assumptions in decision making (WILL sheets). The developed form was attached to the corrective action in the apparent cause evaluation; however, when questioned on the results of the implementation of the form, the licensee could not provide evidence that it had been employed. For another corrective action, the inspectors observed that the licensee had recently identified that not all of the required employees had received the training developed to address the theme, despite the fact that the corrective action had been closed out. Finally, an effectiveness review encompassing the corrective actions in the apparent cause analysis to address the theme has not been performed (currently scheduled completion date is September 2012).

When reviewing the theme in IR 05000255/2011005, the inspectors noted that although the apparent cause centered around decision making being based on minimum regulatory requirements, that a broader look at performance issues resulting from inadequate conservative decision making was more appropriate. At the time, the inspectors noted that the site's recovery plan elements established to address leadership engagement, correction of performance gaps and degradation of safety culture principles more accurately characterized the causes of various performance issues and that actions in the recovery plan would provide a mechanism more likely to achieve sustainable results.

Since the end of 2011, the inspectors have noted continued weakness in the areas of conservative decision making. An additional finding in this area was self-revealed during the first quarter of 2012 when reactor head vents were inadvertently left open (described in IR 05000255/2012002). During the second quarter of 2012, confusion between operations and maintenance personnel resulted in work commencing on the main transformer with the plant on line without the knowledge of the operations shift. In preparation for work to replace both battery breakers during the refueling outage, it was recognized that energized leads may be present but this information was placed in tagout notes vice more clearly in the work instruction. When attempting to establish solid plant pressure control near the end of the refueling outage, the operations shift could not initially maintain control due to a valve that was left out of position. It was believed that a tagout restoration performed on the system would restore the valve to the correct position. Finally, in May 2012 a team of fix-it-now personnel were troubleshooting an issue with a Component Cooling Water valve without the proper oversight or supervisory direction. This exacerbated poor decision-making in the field that eventually led to the control room receiving multiple, unexpected alarms. A momentary ground on a Direct Current (DC) system had been created when Alternating Current power was inadvertently connected to DC power. This issue will be followed up during baseline inspection activities in the third quarter. The substantive cross-cutting issue will be further reviewed as part of the NRC's mid-cycle assessment process.

This review constituted a single selected issue for followup inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.5 Semiannual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. Additionally, the inspectors focused on licensee efforts to address the open cross-cutting aspect theme in Procedures (H.2.c) and progress made to date. The inspectors' review nominally considered the 6 month period of November 2011 through April 2012, although some examples expanded beyond those dates where the scope of the trend warranted. An example of this would be trends in procedure quality which the inspectors assessed through the most recent refueling outage.

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

The inspectors observed some instances of inadequate work package quality. Many inspector observations were during field work in the course of the refueling outage. Such work included replacement of reactor protection system (RPS) M-contactors in each RPS channel with a different model, replacement of containment high pressure RPS pressure switches, and replacement of the DC battery breakers to restore conformance to the design basis (which was the subject of a Yellow finding discussed in IR 05000255/2011020). In each of those activities, the inspectors observed work having to be suspended to address discrepancies in the work packages. Examples of issues encountered were labeling differences between the components and work instruction, a different type of electrical connection on the RPS contactors than expected, different sized instrument tubing than what was required given original work scope, and electrical energy that would be present in a work area that had not been adequately described in the work order. The inspectors believed these issues could have been addressed before work commenced and should have resulted in better procedures. In each case, the inspectors did note positive behavior by the field workers and supervision in that work was stopped, put in a safe condition and re-planned as necessary before continuing on. This was observed as a general trend by the inspectors as well for first and second quarters of 2012 (i.e., increased propensity of workers to stop and fix procedures). For emergent work during the outage, the inspectors noted examples when the resulting procedures lacked the expected detail. Specifically, the inspectors believed steps to address the straightening of the incore instrumentation (ICI) stalk (bent during head installation) and associated replacement of ICI components lacked the necessary detail. Procedural adequacy was also a factor in improper installation of rigging on the reactor head and separately during movement of the head back to the vessel (which resulted in the bent ICI stalk). For this sample, the inspectors also reviewed the backlog of Document Revision Notices (DRNs) and noted an increase in operations and maintenance DRNs since the beginning of 2012 while there were slight

reductions in other areas. While the DRN backlog remains high, it appears the licensee has devoted more resources (contract procedure writers) to address the issue. The inspectors also discussed the status of corrective actions to address the cross-cutting theme with the licensee and reviewed metrics that were used to gauge progress. The inspectors determined corrective actions were being employed as stated in the governing CAP documents and that reasonable metrics were being employed. The inspectors also reviewed a sample of the licensee's Quality Assurance organization's observations regarding procedure quality. The inspectors have discussed their observations with licensee staff and observed corrective actions taken for the individual issues listed above. The cross-cutting theme in Procedure Quality (H.2.c) will be further reviewed by the inspectors as part of the NRC's mid-cycle review process where further actions will be determined.

This review constituted a single semiannual trend inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.6 Institute of Nuclear Power Operations Plant Assessment Report Review

a. Inspection Scope

During the quarter, the inspectors reviewed the Institute of Nuclear Power Operations (INPO) Accreditation Report on Maintenance and Technical Programs prepared for the Palisades site. No issues of concern were identified.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On July 25, 2012, the inspectors presented the inspection results to the Site Vice President, 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.

Further discussion regarding the finding in section 1R08 occurred with your staff on August 3, 2012, after additional information was provided to the NRC.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the inservice inspection were discussed with the Site Vice President, Mr. A. Vitale, on April 27, 2012.

- The inspection results for the areas of Radiological Hazard Assessment and Exposure Controls, Occupational ALARA Planning and Controls, and In-Plant Airborne Radioactivity Control were discussed with Mr. A. Vitale, Site Vice President, on April 20, 2012.

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

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

A. Vitale, Site Vice President
D. Bemis, ISI Program Owner
P. Deeds, Programs Engineer
B. Dotson, Licensing
J. Hager, Steam Generator Program Owner
G. Schrader, Programs Engineering Supervisor
C. Sherman, Radiation Protection Manager

Nuclear Regulatory Commission

John B. Giessner, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000255/2012003-01	NCV	Inadequate Design Margins for Evaluation of Leaking SIRWT Nozzles
05000255/2012003-02	NCV	Operation of Primary Coolant Pumps Outside Design Basis
05000255/2012003-03	NCV	Failure to Follow Work Management Process for Reactor Head Work

Closed

05000255/2012003-01	NCV	Inadequate Design Margins for Evaluation of Leaking SIRWT Nozzles
05000255/2012003-02	NCV	Operation of Primary Coolant Pumps Outside Design Basis
05000255/2012003-03	NCV	Failure to Follow Work Management Process for Reactor Head Work

Discussed
None

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- Admin 4.02, Control of Equipment, Revision 62
- Admin 4.28, Control of Palisades Switchyard Activities, Revision 5
- CR-PLP-2011-04193, Switchyard System has Exceeded Its Maintenance Rule Performance Criteria, August 23, 2011
- CR-PLP-2012-00924, Palisades Review of NRC Event Report for Potential Design Vulnerability to the Bus Undervoltage Scheme at Byron Nuclear Station, February 8, 2012
- CR-PLP-2012-02206, Perform OE Evaluation of NRC-RIS-2011-12-R1, "Adequacy of Station Electric Distribution System Voltages," April 5, 2012
- CR-PLP-2012-03407, EX-10 (Main Transformer) Cooling Lost While in Backfeed Mode, April 29, 2012
- CR-PLP-2012-03471, Startup Transformer Lightening Arrestor Test Data Indicates a Declining Trend, May 1, 2012
- CR-PLP-2012-03971, Received Alarm, EK-0325, "Unit Transformers Trouble," Unexpectedly, May 18, 2012
- CR-PLP-2012-03979, Received Alarm, EK-530, "Startup Transformers Trouble," Unexpectedly, May 19, 2012
- CR-PLP-2012-04184, Found Roosevelt Line Not Updated in EOOS, May 29, 2012
- CR-PLP-2012-04205, Received Alarm, EK-0325, "Unit Transformers Trouble," Unexpectedly, May 30, 2012
- CR-PLP-2012-04398, Trees Below the 345kV Transmission Towers Should Be Cleared, June 8, 2012
- DBD-6.02, 345kV Switchyard, Revision 3
- Generator Interconnection Agreement, January 19, 2011
- ONP-2.1, Loss of AC Power, Revision 14
- RTO-EOP-002-r13, MISO Market Footprint and Sub-area Capacity Emergencies Procedure, February 24, 2012
- RTO-OP-003-r17, Protocols for Nuclear Plant/Electric System Interfaces Procedure, March 23, 2012
- SOP-30, Attachment 6, Station Power System Checklist, Revision 63
- SOP-32, 345 kV Switchyard, Revision 31

1R04 Equipment Alignment

- CR-PLP-2011-05123, Exhaust Leak on Emergency Diesel Generator 1-1, October 6, 2011
- CR-PLP-2011-05347, Incorrect Alignment of Valve Position Indicator and Upper Limit Switch on PCV-6003, October 14, 2011
- CR-PLP-2011-05574, During Performance of MO-7A-2 on K-6B, 1-2 Emergency Diesel Generator, observed erratic behavior of Voltage Regulator in Automatic Mode, October 24, 2011
- CR-PLP-2011-05580, Sparking observed from EDG1-2 Generator Slip Ring Brushes during MO-7A-2, October 24, 2011

- CR-PLP-2012-00387, Severity Level 5 Fuel Oil Leak on Cylinder 7L of K-6B, 1-2 Diesel Generator during performance of RO-128-2, January 16, 2012
- CR-PLP-2012-00454, Fuel Leakage from Cylinder 8L and 9L, January 18, 2012
- CR-PLP-2012-00902, Diesel Generator 1-2 Jacket Water Nitrites have trended down from 976 ppm to 748 ppm, February 8, 2012
- CR-PLP-2012-01208, Incorrect Procedure Guidance in MO-7A-2 and MO-7A-1, February 22, 2012
- CR-PLP-2012-01453, P-905A, D/G 1-1 Pre Lube Oil Pump has an abnormal rubbing/grinding noise, March 5, 2012
- CR-PLP-2012-01778, Unsatisfactory Failure Rate of the New Style Diesel Generator Air Start Pressure Control Valves, March 17, 2012
- SOP-19, Instrument Air System, Attachments 5 and 10, Revision 57
- SOP-22, Emergency Diesel Generators, Revision 54
- M-208, “P&ID Service Water System,” Sheet 1A, Revision 61
- M-208, “P&ID Service Water System,” Sheet 1B, Revision 36
- SOP-15, “Service Water System,” Checklist 15.1, Revision 51

1R05 Fire Protection

- CR-PLP-2012-02717, NRC-Identified Materials in Level 1 Transient Combustible Exclusion Area Without an Approved TCE, April 16, 2012
- CR-PLP-2012-04118, Potential to Lose All Fire Pumps for a Fire in a Given Area Due to Fire Induced Cable Damage, May 24, 2012
- CR-PLP-2012-04187, Issue with the Proximity Switch on Door-44 (Cable Spreading Room Double Door Exit), May 29, 2012
- CR-PLP-2012-04553, Transient Combustible Material Has Been Staged in the Turbine Building 625’ Elevation and the SIRW Tank Roof That Is Not Being Tracked By a Transient Combustible Evaluation, June 18, 2012
- CR-PLP-2012-04553, Transient Combustible Material Staged in Turbine Building 625’ That Is Not Being Tracked By An Evaluation, June 18, 2012
- CR-PLP-2012-04607, Transient Combustible Evaluation Has Not Been Requested for Items on the Turbine Deck in Support of the SIRW Tank Repair, June 20, 2012
- Fire Protection Implementing Procedure 4, Fire Protection Systems and Fire Protection Equipment, Revision 28
- FPSP-SO-4, Fire Suppression Water System Post Indicator valve Operation, Revision 2
- Palisades 1-D Switchgear Room Pre-Fire Plan (Fire Area 3)
- Palisades Cable Spreading Room Pre-Fire Plan (Fire Area 2)
- Palisades Plant Fire Hazards Analysis Report, Revision 7
- Palisades Screenhouse/Intake Structure Pre-Fire Plan (Fire Area 9)
- Palisades Turbine Building / Elev. 607’ to 612’ Pre-Fire Plan (Fire Area 23)
- Palisades Turbine Building / Elev. 625’ Pre-Fire Plan (Fire Area 23)

1R08 Inservice Inspection Activities

- AREVA 03-9175728-000, Palisades Unit 1 Steam Generator Eddy Current Analysis Guidelines, February 1, 2011
- AREVA 51-9142276-000, Technical Justification Report for the ET Surface Examination of J-Groove Welds and Weld Regions, August 12, 2010
- AREVA 51-9179646-000, Steam Generator Degradation Assessment for Palisades 1R22 Inspection, Spring 2012

- AREVA Procedure 54-ISI-460-003, Multi-Frequency Eddy Current Examination of Nozzle Welds and Regions, March 17, 2010
- AREVA Procedure 54-ISI-493-005, Multi-Frequency Rotating Eddy Current Examination of Thick-Walled Tubular Products, February 21, 2011
- AREVA Procedure 54-ISI-604-011, Automated Ultrasonic Examination of Open Tube RPV Closure Head Penetrations, January 25, 2012
- CEP-NDE-0423, Manual Ultrasonic Examination of Austenitic Piping Welds (ASME XI), Revision 5
- CEP-NDE-0496, Manual Ultrasonic Examination of Dissimilar Metal Welds, Revision 5
- CEP-NDE-0641, Liquid Penetrant Examination for ASME Section XI, Revision 7
- CEP-NDE-0955, Visual Examination (VE) of Bare-Metal Surfaces, Revision 303
- CR-PLP-2012-02604, Scaffolding Was Built to the Wrong Train for Containment Feedwater Pipe Support Inservice Inspection (ISI), April 14, 2012
- CR-PLP-2012-02608, Clerical and Administrative Errors Were Identified in Three 2009 and 2010 Inservice Inspection (ISI) Reports, April 14, 2012
- CR-PLP-2012-02674, During the 1R22 Scheduled ISI Examination of Safety Injection System Component ESS-6-SIS-1B1-12 (Point 121) A Ventilation Ductwork Flange was identified as Being In contact With the SIS Piping Adjacent to the Examination Location April 15, 2012
- CR-PLP-2012-02813, Qualified PDI Examination of Two Cold Leg Drain Line Dissimilar Metal Welds Cannot Be Performed Due to Piping Geometry, April 17, 2012
- CR-PLP-2012-02910, ISI Inspection Item No. 4, Pie Restraint ESS-2-SIS-1B1-7PR(H-780) Vertical Hanger Rod Contacts a Horizontal Small Bore Pipe or Conduit, April 19, 2012
- CR-PLP-2012-03344, 2012 Bare Metal Visual Examination of the Reactor Pressure Vessel Head Relevant Indications Identified, May 28, 2012
- EN-DC-319, Inspection and Evaluation of Boric Acid Leaks, Revision 8
- SEP-BAC-PLP-001, Boric Acid Corrosion Control Program, Revision 0
- SG-SGMP-11-1, Cycle 22 Steam Generator Operational Assessment Report, January 20, 2011

1R11 Licensed Operator Requalification Program

- EI-1, “Emergency Classification and Actions,” Revision 49
- EOP-1.0, “Standard Post-Trip Actions,” Revision 13
- EOP-4.0, “Loss-of-coolant Accident Recovery,” Revision 20
- Simulator Scenario PLSEG-LOR-12B-03, Revision 0
- SOP-1B, “Primary Coolant System – Cooldown,” Revision 13
- SOP-1C, “Primary Coolant System – Heatup,” Revision 13

1R12 Maintenance Effectiveness

- EGAD-EP-10, Maintenance Rule Scoping Document, Revision 5
- Work Order Summary report for HAC and SPS, report run May 17, 2012
- VLF/Tan-D testing results for various 4160V cables tested 1R22
- CR-PLP-2012-03375, EOC for degraded medium voltage cables, April 29, 2012
- Plant Cable Reliability Program In-Scope Cable Listing
- WO 51619206, EX-05 Load Cable Testing
- PLP-RPT-10-00029, Review of Periodic Inspection and Testing of Inaccessible Medium Voltage Cables and Manholes in Scope of License Renewal Aging Management Program
- FSAR Chapter 1, Introduction and General Plant Description, Revision 29
- EN-DC-346, Cable Reliability Program, Revision 3
- CR-PLP-2012-03161, Failed cables during Tan-D testing, April 24, 2012

- EN-MA-138, VLF Tan-D and Withstand Testing of Electrical Power Cables, Revision 1
- CR-PLP-2012-03657, 252-302 Both charging motor fuses blew during PMT, May 6, 2012
- 2010 and 2011 System Health Reports for 4160V AC System

1R13 Maintenance Risk Assessments and Emergent Work Control

- Component/Plant Impact for Work Order 00316803-11,
- CR-PLP-2012-02679, S/G Nozzle Dam Consoles Have a Single Point of Failure Issue, April 15, 2012
- CR-PLP-2012-02763, Rising Containment Sump Fill Rate Due to Leakage on Loop 1A Nozzle Dam on ‘A’ S/G, April 17, 2012
- CR-PLP-2012-02766, S/G E-50A Cold Leg Loop A Wet Nozzle Dam Seal Leakage Detected on Nozzle Console, April 17, 2012
- CR-PLP-2012-02770, S/G E-50A Hot Leg Wet Nozzle Dam Seal Leakage Detected on Nozzle Console, April 17, 2012
- CR-PLP-2012-03486, Improvements for Reactor Head Setting Procedure, May 1, 2012
- CR-PLP-3411, During Setting the Reactor Head on the Vessel Flange, Damage Occurred to the ICI Flange #6, April 29, 2012
- EC 37034, “Analysis of the Lateral Force Required to Straighten ICI Stalk #6 During 1R22 Refueling Outage,” Revision 0
- GOP-14, Shutdown Cooling, Attachment 3, Shutdown Cooling Equipment Availability June 14- 17 2012
- RFL-D-20, “Replace Reactor Vessel Closure Head Metal O-Rings,” Revision 3
- RFL-R-16, “Reactor Vessel Closure Head Installation,” Revision 9
- RFL-SG-2, “S/G Primary Nozzle Dam Installation and Removal,” Revision 6
- WO 313672, “ICI Stalk Associated with Flange 6 is Bent,” April 30, 2012
- WO 52326102, “Reactor Disassembly, Reassembly, and Fuel Moves (1R22),” April 30, 2012
- RO-217, TS Leakrate Testing of Engineered Safeguards Check Valves, Revision 4
- RT-71L, TS Admin 5.5.2 Pressure Test of ESS Pump Suction Piping, Revision 6
- EN-LI-100, Process Applicability Determination, Revision 11
- WO-314139, RO-217 Partial Sec 5.5 Leak Rate Test
- GL-88-17, Loss of Decay Heat Removal
- FSAR Chapter 5, Design of Structures, Systems, and Components, Revision 24
- FSAR Section 14.20, Liquid Radwaste Incident, Revision 29
- FSAR Chapter 11, Radioactive Waste Management and Radiation Protection, Revision 27
- EN-DC-115, Engineering Change Process, Revision 13
- EN-DC-136, Temporary Modifications, Revision 7

1R15 Operability Determinations and Functionality Assessments

- CR-PLP-2012-03497, Ultrasonic readings out of tolerance for stud elongation, May 2, 2012
- RFL-SG-1, Primary Manway Cover Removal and Installation, Revision 2
- ASME Code Case N-705, Evaluation Criteria for Temporary Acceptance of Degradation in Moderate Energy Class 2 or 3 Vessels and Tanks, October 12, 2006
- PLP-RPT-12-00097, Evaluation of SIRW Tank T-58, Revision 0 and 1
- CR-PLP-2012-03368, Leakage found in SIRWT catacombs, April 29, 2012
- CR-PLP-2012-03551, Resident inspector questioned method used to remove N-nozzle from service, May 3, 2012
- CR-PLP-2012-03691, Operability evaluation references incorrect critical flaw size, May 6, 2012

- CR-PLP-2012-03732, NRC has raised questions about adequacy of technical content in SIRWT evaluation, May 8, 2012
- CR-PLP-2012-04245, Previous condition report does not address potential performance deficiency associated with lack of detail in PLP-RPT-12-00097, May 31, 2012
- CR-PLP-2012-04246, Increasing trend in SIRWT leakage, May 31, 2012
- ODMI for T-58 Leakage, Revision 2
- CR-PLP-2007-04585, Large metallic debris found in reactor vessel, September 27, 2007
- Flowserve letter to Palisades regarding primary coolant pump debris, September 28, 2007
- FSAR Chapter 4, Primary Coolant System, Revision 28
- CR-PLP-2007-01175, P-50D High priority single point vulnerability, March 14, 2007
- CR-PLP-2010-06576, Inspection of P-50D indicates conclusions of 2007 need to be revisited, December 14, 2010
- CR-PLP-2011-05744, P-50C experienced change in vibration, October 29, 2011
- Flowserve. (April 9, 2012). "CFD Analysis on Palisades Primary Circulation Pump Impeller Damage," EC36319 PLP-12-00085
- "Primary Coolant Pump P-50C Loose Impeller Piece Travel Path to Cold Leg FTDs," PLP-RPT-12-00093, May 2, 2012
- CR-PLP-2012-06427, Historical trend of impeller vane cracks, November 23, 2011
- CR-PLP-2001-01190, Cracks Identified in Old Primary Coolant Pump P-50B Impeller, October 11, 2001
- Services, M. C. Operational Assessment of Primary Coolant Pump, P-50C, Entergy Engineering Report No. PLP-RPT-11-00029 Rev. 0., December 14, 2011
- Westinghouse Evaluation, Disposition of Postulated Foreign Material in Palisades Muclear Plant Nuclear Steam Supply System Originating from a Primary Coolant Pump Impeller, DAR-SEE-11-12-6, April, 2012

<u>1R18 Plant Modifications</u>

- CR-PLP-2012-03156, While Walking Down the Temporary Service Water Discharge Piping in the Turbine Building, A Leak Was Noticed Coming From the Piping Above the East Mezzanine, April 24, 2012
- CR-PLP-2012-03160, Upon Closure of Line-Stop on the Critical Service Water Header, Leakage Was Identified At A Piping Opening on the Non-Critical Service Water Header in the Turbine Building, April 24, 2012
- Design Basis Document 1.01, “Component Cooling Water System,” Revision 2
- Design Basis Document 1.02, “Service Water System,” Revision 2
- EA-EC10749-01, “Component Cooling Water and Spent Fuel Pool Heat Exchanger Performance During EC34191,” April 25, 2012
- EC 34191, “Temporary Discharge Pipe Service Water Installation (1R22),” March 2, 2012
- EC 36899, “Temporary Modification Change Notice to EC34191: Allow For A Lower Minimum SW Flowrate Based On Current Plant Conditions (Mode 6)”
- M-208, “P&ID Service Water System,” Sheet 1A, Revision 59
- PAD #12-0091, “EC34191, Temporary Service Water Discharge Pipe Installation (1R22),” Revision 1
- Service Water Repair Plan for CV-0824, Service Water Return from Containment
- SOP-16, Attachment 4, “Align Service Water System Returns Via EC34191 Temporary Discharge Pipe,” Revision 35
- WO 303191, “Install Temp Mod for SW Piping Alternate Discharge per EC 34191,” April 20, 2012
- EC 34131, Battery Breaker Replacements
- EN-WM-107, Post-Maintenance Testing, Revision 4

- RE-83A, Service Test, Battery ED-01, Revision 20
- WO-291601, 72-01, Replace Breaker Assembly
- EN-OP-119, Protected Train Postings, Revision 4

1R19 Post-Maintenance Testing

- CR-PLP-2012-02348, Wire loose on M1 Contactor circuitry, April 10, 2012
- CR-PLP-2012-02416, While Performing Inspection/Cal of ED-06 the Bypass Source AC Input Meter Was Reading Voltage with the Breaker Open, April 11, 2012
- CR-PLP-2012-02863, Two Critical Steps Were Not Completed As Written in WO 292954-01 for CV-1059, April 18, 2012
- CR-PLP-2012-02898, During Disassembly of the Operator for CV-0510 a Degraded O-Ring Was Found, April 19, 2012
- CR-PLP-2012-02917, Repack of CV-1057 Showed Valve Stem and Part of Stuffing Box Corroded, April 19, 2012
- CR-PLP-2012-03171, Thermal Light is on for MO-0501, MSIV Bypass, April 24, 2012
- CR-PLP-2012-03218, When Reassembling CV-0510, the Lower Stud Spherical Washers Were Larger Than the Recess Bore, April 25, 2012
- CR-PLP-2012-03334, Needed to Machine the Disc Arm for CV-0501, April 28, 2012
- CR-PLP-2012-03409, As-Found Condition of CV-0501 Illustrates the Need to Repair Valve, April 29, 2012
- CR-PLP-2012-03455, During As-Found Inspections of CV-0501, Pitting Was Discovered on Bearing End Gasket Sealing Surface of Valve Body, May 1, 2012
- CR-PLP-2012-03494, WO 52325621 for CV-0501 Showed Disc Assembly Has Severe Wear and Degradation on Various Parts, May 2, 2012
- CR-PLP-2012-03629, During RO-19, CRD-1 Did Not Stop At Its Lower Electrical Limit (LEL), May 5, 2012
- CR-PLP-2012-03639, During RO-19, SPI (Secondary Position Indication) for CRD-6 Did Not Change Reading, May 5, 2012
- CR-PLP-2012-03640, During RO-19, Part Length Control Rod (CRD-43) Would Not Move, May 5, 2012
- CR-PLP-2012-03649, Work Order Performed for MO-0510 (MSIV Bypass) Found Lug Shorted, May 5, 2012
- CR-PLP-2012-03658, While Performing QO-37 for CV-0510 Would Not Fully Close Per Local Measurement, May 6, 2012
- CR-PLP-2012-3015, During Disassembly of CV-0510 the Belleville Washers on the Upper Disc Retaining Studs were Broken and Pieces Were Missing, April 21, 2012
- CS-OP-PR-075, Operation of the SIRW Tank Process Equipment at Entergy-Palisdes Nuclear Station, Revision 0
- EC 27671, Replace RPS M-Contactors
- EC 31670, PS-1801 Replacement
- EC 36792, “Temporary Pipe Plug in Main Steam Line Upstream from CV-0510 & CV-0501,” Revision 0 (and Attachments)
- EC 37206, “Evaluation of QO-37 Close Measurement for CV-0510”
- EN-OP-116, Infrequently Performed Tests or Evolutions, rev. 009
- MSM-M-57, “Permanent Maintenance Procedure: Universal Diagnostic System Operating Procedure,” Revision 9
- Process Applicability Determination #2012-0166, “EC 36792: Temporary Pipe Plug in Main Steam Line Upstream from CV-0510 & CV-0501,” Revision 0
- QI-3, Reactor Protection Matrix Logic Tests, Revision 5
- QO-37, “Main Steam Isolation and Bypass Valve Testing,” Revision 7

- RO-19, “Control Rod Position Verification,” Revision 24
- SK-06152010, SIWR (SIC) Tank Process Equipment P&ID, Revision A
- Survey PLP-1206-0055, Frac tank Survey, June 18, 2012
- WI-EPS-E-02, “Inverter Maintenance,” Revision 11
- WI-EPS-E-04, “Calibration Testing of Electrical Meters,” Revision 1
- WO 244626, PS-1801, Replace PS-1801
- WO 292954, “CV-1059 Pressurizer Spray Valve Repack and Testing,” April 17, 2012
- WO 293278, “CV-1057 Pressurizer Spray Valve Repack and Testing,” April 17, 2012
- WO 52325527, RI-6B, Containment Pressure Channel Calibrations-Finals
- WO 52325621, “CV-0501: Disassemble, Inspect, and Reassemble”
- WO 52325622, “CV-0510: Disassemble, Inspect, and Reassemble”
- WO 52325748-02, “ED-06 Inverter #1 PM,” April 10, 2012
- WO-264735, Replace RPS Contactor

1R20   Outage Activities

- CLP-M-6, Inspection of Heavy Load Devices, Revision 13
- CR-PLP-2012-02610, Reactor Vessel Head Detensioning Issues, April 14, 2012
- CR-PLP-2012-02848, Partial retensioning reactor head not meeting station standards and expectations, April 18, 2012
- CR-PLP-2012-03141, During PWR Core Loading Verification, A Piece of Debris was Discovered at Core Location T-8, April 24, 2012
- CR-PLP-2012-03587, Water Became Entrained in Temporary Hoses Used in PCS Vacuum Fill, May 4, 2012
- CR-PLP-2012-03595, During PCS Fill Encountered Transmitter Errors Due to Aux Spray Lineup, May 4, 2012
- CR-PLP-2012-03784, T-399 Could Not Be Completed Satisfactorily, May 9, 2012
- CR-PLP-2012-03804, During Low Power Physics Testing, CRD-1 White Light Did Not Come On, May 11, 2012
- CR-PLP-2012-03831, P-1B, Main Feedwater Pump, Experiencing Severe Oscillations, May 12, 2012
- EN-FAP-OM-006, “Working Hour Limits for Non-Covered Workers,” Revision 4
- EN-FAP-OU-108, “Fuel Handling Process,” Revision 3
- EN-IS-124, Industrial Safety Planning and Job Safety Hazards Analysis, Revision 4
- EN-MA-118, “Foreign Material Exclusion,” Revision 8
- EN-MA-118, Foreign Material Exclusion, Revision 8
- EN-OM-123, “Fatigue Management Program,” Revision 3
- EN-OU-108, “Shutdown Safety Management Program (SSMP),” Revision 3
- EN-RE-326, “PWR Core Loading Verification,” Revision 0
- EN-WM-102, Work Implementation and Closeout, Revision 7
- EOP Supplement 1, “Pressure Temperature Limit Curves,” Revision 5
- GOP-11, “Refueling Operations and Fuel Handling,” Revision 45
- GOP-14, “Shutdown Cooling Operations,” Revision 44
- GOP-2, “Mode 5 to Mode 3 ≥525°F,” Revision 34
- GOP-3, “Mode 3 ≥525°F to Mode 2,” Revision 31
- MSM-M-71, “Containment Cleanliness Implementation Plan and Containment Closeout,” Revision 10
- ONP-17, “Loss of Shutdown Cooling,” Revision 41
- ONP-23.3, “Loss of Refueling Water Accident,” Revision 6
- ONP-23.4, “Loss of Spent Fuel Pool Cooling,” Revision 3
- PO-2, “PCS Heatup/Cooldown Operations,” Revision 5

- RFL-D-15, “Instrument Nozzle (ICI) Flange Removal,” Revision 4
- RFL-D-16, “Reactor Vessel Closure Head Removal,” Revision 10
- RFL-D-18, “Remove Incores and RVLMS Detectors from UGS,” Revision 3
- RFL-D-20, “Replace Reactor Vessel Closure Head Metal O-Rings,” Revision 3
- RFL-D-3, Open Equipment Hatch, Revision 4
- RFL-R-13, Reactor Pressure Vessel Detensioning, Revision 3
- RFL-R-16, “Reactor Vessel Closure Head Installation,” Revision 9
- RFL-R-18, “Install Incore and RVLMS Detectors into UGS,” Revision 5
- RFL-SG-2, “S/G Primary Nozzle Dam Installation and Removal,” Revision 6
- RFL-V-9, “Core Mapping System Setup and Operation,” Revision 4
- RT-191, “Startup Physics Test Program,” Revision 8
- RT-92, Inspection of Containment Sump Envelope, Revision 5
- SOP-1A, “Primary Coolant System,” Revision 18
- SOP-1B, “Primary Coolant System – Cooldown,” Revision 13
- SOP-1C, “Primary Coolant System – Heatup,” Revision 13
- SOP-2A, “Chemical and Volume Control System,” Revision 74
- SOP-3, “Safety Injection and Shutdown Cooling System,” Revision 84
- T-399, “Pressurizer Backup Heater Test,” Revision 0
- Tagout 1R22-1 EDC-001-72-01
- WI-PCS-M-06, “NSSS Walkdown,” Revision 3
- WO 244482, Contingency Work Order for Stuck Reactor Vessel Nut

1R22 Surveillance Testing

- CR-PLP-2011-00062, K-6A Emergency Diesel Generator Petlock Valve Vibrations, January 5, 2011
- CR-PLP-2011-01556, CRD-21 Secondary Rod Position Indication (SPI) Inoperable, March 30, 2011
- CR-PLP-2011-02350, P-8B Steam Driven Auxiliary Feedwater Pump tripped on Overspeed during Surveillance Testing, May 10, 2011
- CR-PLP-2011-04068, Surveillance step 12.0 of MO-7A-2, 1-2 Diesel Generator Surveillance test not performed, August 18, 2011
- CR-PLP-2011-04361, Control Room HVAC Heat Removal Capability Surveillance not completed, September 1, 2011
- CR-PLP-2012-00886, Tech Spec Surveillance RV-0401 Frequency, February 7, 2012
- EC 20930, Peak Containment Pressure, Revision 0
- RO-32-65, LLRT – Local Leak Rate Test Procedure for Penetration MZ-65, Revision 14
- CR-PLP-2012-02408, RO-112 Did Not Meet Its Acceptance Criteria for PRV-1067/1068 and PCV-1069/1070, April 11, 2012
- CR-PLP-2012-02659, GL-2008-01 Gas Void Monitoring Point #22, HPSI Hot Leg 1 Injection, Was Found Partially Void of Water During 1R22, April 15, 2012
- CR-PLP-2012-02933, During RO-216 the Total Flow to Containment Acceptance Criteria Was Not Met, April 20, 2012
- CR-PLP-2012-02939, FI-1772, Containment Air Cooler VHX-3 Flow Found “Stuck” at 680 gpm When Performing RO-216, April 20, 2012
- CR-PLP-2012-02958, During RO-216, VHX-1 (Containment Air Cooler) Was Out of the 10% Deviation, April 20, 2012
- CR-PLP-2012-03000, DAS Failed to Collect Accurate Data During RT-8D, April 21, 2012
- CR-PLP-2012-03028, During Post-Cal of Instruments for RO-216 As-Found Settings for FI-1771 (Containment Air Cooler VHX-2 Flow) Were Out of Tolerance High, April 21, 2012

- CR-PLP-2012-03031, During Post-Cal of Instruments for RO-216 As-Found Settings for FI-0823 (Component Cooling HX E-54A Service Water Flow) Were Out of Tolerance High, April 21, 2012
- CR-PLP-2012-03033, During Post-Cal of Instruments for RO-216 As-Found Settings for FI-1772 (Containment Air Cooler VHX-3 Flow) Were Out of Tolerance High, April 21, 2012
- CR-PLP-2012-03042, Charging Pump P-55C Did Not Start During RT-8C, April 22, 2012
- CR-PLP-2012-03044, Issues Identified During Performance of RT-8C, April 22, 2012
- CR-PLP-2012-03047, Wrong Method of Measurement Was Used to Determine Leakage in RO-216, April 22, 2012
- CR-PLP-2012-03058, During Post-Cal of Instruments for RO-216 As-Found Settings for FT-0883 (Containment Air Cooler Service Water Leak Detector) Were Out of Tolerance, April 22, 2012
- CR-PLP-2012-03079, During the Performance of TSST RT-8D, HPSI P-66A Tripped Unexpectedly, April 22, 2012
- CR-PLP-2012-03085, DAS Failed to Obtain Data for Normal Shutdown Sequencer During Both Performances of RT-8D, April 23, 2012
- CR-PLP-2012-03086, During TSST RT-8D Breaker 152-207 (HPSI P-66A Supply) Tripped Immediately After Receiving Closed Signal, April 23, 2012
- CR-PLP-2012-03092, While Performing WO for X Phase of HPSI P-66A Relay 150/151-207, the Time Over Current Portion Failed As-Found Settings, April 23, 2012
- CR-PLP-2012-03099, While Performing WO for X Phase of HPSI P-66A Relay 150/151-207, the Time Over Current Setting on Setting Sheet Was Not Correct, April 23, 2012
- CR-PLP-2012-03689, During RO-112 PRV-1067 and PRV-1071 Did Not Meet Acceptance Criteria for Flow Verification, May 7, 2012
- EC 37083, “Engineering Reply for Disposition of RT-8C Testing Results During 1R22,” May 3, 2012
- EC 37084, “Engineering Reply for Disposition of RT-8D Testing Results During 1R22,” May 5, 2012
- QO-14B, “Inservice Test Procedure – Service Water Pumps,” Revision 34
- QO-6, “Cold Shutdown Valve Test Procedure,” Revision 44
- RO-105, “Full Flow Test For SIT Check Valves and PCS Loop Check Valves,” Revision 10
- RO-112, “Reactor Head/Pressurizer Vent Flow Check,” Revision 8
- RO-216, “Service Water Flow Verification,” Revision 15
- 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 26
- RT-8C, “Engineered Safeguards System – Left Channel,” Revision 31
- RT-8D, “Engineered Safeguards System – Right Channel,” Revision 31
- WO 312984-01, “MC-34R101: Test Right Channel NSD Sequencer,” May 2, 2012
- WO 313646-01, “P-55C (‘C’ Charging Pump) Gap Testing,” May 1, 2012

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

- EN-MA-125, Troubleshooting Control of Maintenance Activities, Revision 9
- EN-RP-101, Access Control for Radiologically Controlled Areas, Revision 6
- EN-RP-122, Alpha Monitoring, Revision 6
- EN-RP-503, Selection, Issue, and Use of Respiratory Equipment, Revision 5
- EN-RP-504, Breathing Air, Revision 1
- Radiological Survey, PLP-1204-0231, April 16, 2012
- RFL-D-18, Remove Incores and RVLMS Detectors from UGS, Revision 3
- Work Order 52326068 01, Troubleshoot and Repair All Fuel Handling Equipment, April 16, 2012

2RS2 Occupational ALARA Planning and Controls (71124.02)

- Radiation Work Permit and Associated ALARA Files, RWP 20120431, Refuel Project: Rx and Spent Fuel Pool Tilt Pit Activities, Revision 00
- Radiation Work Permit and Associated ALARA Files, RWP 20120432, Refuel Project: Rx Cavity Decon and Support Activities, Revision 00
- Radiation Work Permit and Associated ALARA Files, RWP 20120433, Refuel Project: Rx Vessel Disassembly, Revision 01
- Radiation Work Permit and Associated ALARA Files, RWP 20120454, S/G Primary Side Activities, Revision 04
- Radiation Work Permit and Associated ALARA Files, RWP 20120468, Repack Pressurizer Spray Controls CV-1057 and CV-1059, Revision 00

2RS3 In-Plant Airborne Radioactivity Control and Mitigation (71124.03)

- CR-PLP-2012-02683, Radiation Protection Airborne Radioactivity Air Samples Are Not Being Analyzed In a Timely Manner. Delays Can Impact RP Job Coverage for Dose Assessment, April 15, 2012
- EN-RP-131, Air Sampling, Revision 9
- EN-RP-131-"Air Sampling," Attachment 9.9, Tritium/Transuranic DAC-Hour Data Collection Sheet, DH Calc No.4739561
- EN-RP-131-"Air Sampling,"Attachment 9.9, Tritium/Transuranic DAC-Hour Data Collection Sheet, DH Calc No. 4739561a
- WI-RSD-H-014, Air Sample Spreadsheet, Revision 9

4OA1 Performance Indicator Verification

- Selected logs, April 1 2011 through March 31, 2012

4OA2 Problem Identification and Resolution

- CR-PLP-2011-03577, The Station Received Its Third NRC Violation with a Cross-Cutting Aspect in Management Oversight, July 20, 2011
- CR-PLP-2011-04522, There are a Total of 5 Cross-Cutting Aspects in the Area of H.4.c for this Assessment Period, September 12, 2011
- LO-WTPLP-2011-00366 CA 785, "Common Cause Analysis for Period of January 1, 2011, to December 12, 2011, to Review Organizational and Programmatic Weaknesses," Revision 2, April 12, 2012
- Palisades H4c Management Oversight Cross-Cutting Aspect Briefing Document, May 23, 2012
- Palisades Training Materials for Managers and Supervisors
- Root Cause Evaluation, "Potential Cross-Cutting Violation in Human Performance – Management Oversight (H.4(c))," Revision C, October 13, 2011

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CR	Condition Report
DC	Direct Currrent
DRN	Document Revision Notice
ET	Eddy Current
EPRI	Electric Power Research Institute
gpd	Gallons Per Day
ICI	Incore Instrumentation
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
LOCA	Loss-of-Coolant Accident
NCV	Non-Cited Violation
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OSP	Outage Safety Plan
PARS	Publicly Available Records System
PCP	Primary Coolant Pump
PCS	Primary Coolant System
RFO	Refueling Outage
RPS	Reactor Protection System
RWP	Radiation Work Permit
SDP	Significance Determination Process
SG	Steam Generator
SIRWT	Safety Injection and Refueling Water Tank
SSCs	Structures, Systems and Components
TS	Technical Specification
TSO	Transmission System Operator
UFSAR	Updated Final Safety Analysis Report
WO	Work Order

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Sincerely,

/RA by Jay A. Lennartz for/

John B. Giessner, Chief
Branch 4
Division of Reactor Projects

Docket No. 50-255
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Letter to A. Vitale from J. Giessner dated August 8, 2012.

SUBJECT: PALISADES NUCLEAR PLANT INTEGRATED INSPECTION REPORT 05000255/2012003

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