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

May 14, 2013

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 05000255/2013002

Dear Mr. Vitale:

On March 31, 2013, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palisades Nuclear Plant. The enclosed report documents the results of this inspection, which were discussed on April 22, 2013, with yourself and other members of your staff.

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

Based on the results of this inspection, two self-revealed and four 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 this NCV, 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/

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

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

Enclosure: Inspection Report 05000255/2013002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-255
License No:	DPR-20
Report No:	05000255/2013002
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	January 1, 2013, through March 31, 2013
Inspectors:	T. Taylor, Senior Resident Inspector A. Scarbeary, Resident Inspector D. Alley, Senior Materials Engineer M. Jones Jr., Reactor Inspector J. Lennartz, Project Engineer M. Phalen, Senior Health Physicist E. Sanchez-Santiago, Reactor Inspector S. Sheldon, Senior Reactor Engineer R. K. Walton, Senior Operator Licensing Examiner C. Zoia, Operator Licensing Examiner
Approved by:	John B. Giessner, Chief Branch 4 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000255/2013002; 01/01/2013 – 03/31/2013; Palisades Nuclear Plant; Heat Sink Performance; Plant Modifications; Surveillance Testing; Occupational Dose Assessment; Problem Identification and Resolution

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Four Green findings were identified by the inspectors and two Green findings were self-revealed. The findings were considered non-cited violations (NCVs) of NRC regulations. The significance of inspection findings are indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated January 28, 2013. 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. A self-revealing finding of very low safety significance (Green) with associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, and Technical Specification (TS) 3.4.13, Primary Coolant System (PCS) Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of Control Rod Drive Mechanism (CRDM) cracking and leakage, a significant condition adverse to quality (SCAQ). Specifically, for Criterion XVI the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21, caused by transgranular stress corrosion cracking (TGSCC). Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement. The licensee replaced CRDM-24 upper housing and entered the issue into their corrective action program as CR-PLP-2013-01134. Additional corrective actions are described in NRC Inspection Report 05000255/2012012.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Initiating Events Cornerstone attribute of equipment performance and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. The inspectors determined this finding was of very low safety significance (Green) because the leak would not have exceeded the reactor coolant system leak rate for a small LOCA and could not have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Specifically, the slow rate of change for leakage for TGSCC in type 316 stainless steel will experience leakage rates well below a small break LOCA,

which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The cause of this finding, non-conservative decision making, occurred over 10 years ago and is well outside of the nominal 3 year period in IMC 0612 for cross-cutting aspects. Therefore, this is not indicative of current performance, because no other opportunities to identify the issue occurred during the previous 3-year period. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.4 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding. (Section 4OA2.4(b.1))

- Green. The inspectors identified a finding of very low safety significance (Green) with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the 2012 cracking identified in CRDM-24 in accordance with a quality procedure, Procedure, EN-LI-118, "Root Cause Evaluation." This issue was entered into the licensee's Corrective Action Program (CAP) under CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

  The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because if left uncorrected, the performance deficiency would have the potential to lead to a more significant safety concern. Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2, "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 609, Attachment 4, "Initial Characterization of Findings," the inspectors determined that the finding was associated with the Initiating Events Cornerstone because the failure of a CRDM housing is a Primary System LOCA initiator contributor. Using Exhibit 1, "Initiating Events Screening Questions," in IMC 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power," the inspectors determined this finding was of very low safety significance because the leak would not exceed the reactor coolant system leak rate for a small LOCA and would not have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. Specifically, the slow rate of change for leakage for TGSCC in type 316 stainless steel will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture. The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the root cause report (RCR) related to the decision making cross-cutting component in the human performance area because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC when there was not enough information to exclude them from consideration. (H.1(b)). (Section 4OA2.4(b.2))

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control", for failure to establish testing to demonstrate the safety-related Component Cooling Water (CCW) heat exchangers would perform satisfactorily in service. Specifically, the licensee failed to demonstrate the heat exchanger's fouling factors would remain acceptable to ensure adequate heat transfer capability prior to changing the inspection, cleaning, eddy current testing, and thermal performance testing frequency to 12 years. The licensee entered this issue into their CAP as CR-PLP-2012-05132 and CR-PLP-2013-00544 and implemented actions to revise the inspection, cleaning, testing, and maintenance frequencies to less than 5 years.

  The issue was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability reliability and capability of systems needed to respond to initiating events to prevent undesired consequences. Specifically, the inappropriate test frequency affected the licensees' ability to ensure the CCW heat exchangers were available and capable to reliably perform as expected. The finding screened as of very low safety significance (Green) because the inadequate test program was not a design deficiency and did not result in a loss of system or component function. This finding has a cross-cutting aspect in the area of human performance, decision making because the licensee did not use conservative decision making and did not conduct effectiveness reviews of safety significant decisions to verify the validity of underlying assumptions, identify possible unintended consequences, or determine how to improve future decisions. Specifically, the licensee failed to use conservative decision-making or verify the validity of underlying assumptions when evaluating the effect that reducing the frequency of testing, inspection, cleaning, and maintenance would have on the CCW heat exchangers (H.1(b)). (Section 1R07)

- Green. The inspectors identified a finding of very low safety significance (Green) with an associated NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly plan and document work on the safety-related 'A' CCW heat exchanger during a forced outage to repair leaks in the heat exchanger. Contrary to Criterion V and site implementing procedures EN-DC-115, Engineering Change Process, and EN-WM-105, Planning, the licensee did not ensure that appropriate quantitative or qualitative acceptance criteria for determining that important activities affecting quality were included in the work done to re-plug a population of leaking tubes in the heat exchanger. The licensee changed the work instructions to include the acceptance criteria after questioning by the inspectors. The licensee also interviewed workers to ensure the criteria had been utilized during earlier plug installation. The licensee entered the issue into their CAP as CR-PLP-2013-00773 and CR-PLP-2013-00969.

  The issue was determined to be greater-than-minor per IMC 0612, Appendix B, "Issue Screening," because if left uncorrected, it could lead to a more significant safety concern. The inspectors' decision was informed by examples 3j and 3k in IMC 0612, Appendix E, "Examples of Minor Issues." The examples refer to an issue not being minor if significant programmatic deficiencies were identified with the issue that could lead to worse errors if left uncorrected. When the issue was first raised by the inspectors, only one of the two critical parameters was initially added to the revised work

instructions. Further, two examples of inadequate documentation were identified. A basis for removing steps to check for leaks was not properly documented; and it was not clear from the completed work packages that the engineering acceptance criteria were met. Given these issues, the inspectors determined the threshold for a finding was met. The inspectors concluded the finding adversely impacted the Mitigating Systems Cornerstone objective and was of very low safety significance (Green) utilizing IMC 0609, "Significance Determination Process." Specifically, utilizing Exhibit 2 of Appendix A, all questions in Section A were answered 'no'. The finding had an associated cross-cutting aspect in the work control component of the human performance area. Specifically, the licensee did not coordinate work activities by incorporating actions to ensure interdepartmental alignments were made while planning and executing the work to assure plant and human performance (H.3(b)). (Section 1R18)

Green. A self-revealed finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion V was identified for the failure to conduct the 'A' Auxiliary Feedwater (AFW) pump technical specification surveillance test in accordance with the prescribed in-service test procedure. Specifically, plant personnel conducting the surveillance test on the 'A' AFW Pump adjusted packing when it was not required per the guidance in the procedure, which caused the pump packing to overheat and start smoking, resulting in unplanned inoperability of the pump. The licensee documented the issue in their corrective action program as CR-PLP-2013-01128 and completed an apparent cause evaluation. Planned corrective actions included revising the in-service test procedure.

The finding was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," because it was associated with the Mitigating Systems Cornerstone attribute of human performance and adversely impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, a packing adjustment was made without being required by the procedure, causing the pump to overheat, which resulted in unplanned inoperability of the safety-related and risk significant 'A' AFW pump. The finding had an associated cross-cutting aspect in the area of human performance related to the cross-cutting component of resources, in that the licensee ensures plant personnel have complete, accurate, and up-to-date design documentation, procedures, and work packages. In this finding, the fact that the 'A' AFW pump has a unique packing design was not evident in the procedure being used and was not discussed during the pre-job briefs (H.2(c)). (Section 1R22)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of TS 5.4.1. Specifically, the licensee failed to perform Derived Air Concentration (DAC)-Hour tracking for airborne transuranic radioactivity as required by a quality plant procedure, EN-RP-131, "Air Sampling," resulting in untimely internal dose assessments for selected plant workers. The issue was entered in the licensee's corrective action program as CR-PLP-2012-02683. The licensee's immediate corrective actions included re-evaluating the use of site-specific work instructions. Long-term corrective actions included procedure changes and completing the required personnel dose assessments utilizing upper bounding radiological conditions.

The finding is more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. Specifically, not performing DAC-Hour tracking for airborne transuranic radioactivity affected the licensee's ability to assess workers internal exposures in a timely manner and adversely impacted the licensee's ability to monitor, control and limit workers' radiation exposures (committed effective dose equivalent or internal dose). In accordance with IMC 0609 Appendix C, "Occupational Radiation Safety Significance Determination Process," the inspectors determined that the finding had very low safety significance (Green) because the finding: (1) did not involve as-low-as-is-reasonably-achievable (ALARA) planning and controls; (2) did not involve a radiological overexposure; (3) there was not a substantial potential for an overexposure; and (4) there was no compromised ability to assess dose. The inspectors determined that the primary cause of this finding was related to a cross-cutting aspect in the area of human performance, resources component, such that the licensee maintains complete, accurate and up-to-date procedures and work packages. (H.2(c)). (Section 2RS4)

B. Licensee-Identified Violations

Violations of very low safety significance that were identified by the licensee have been reviewed by inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee's corrective action program. These violations and corrective action tracking numbers are listed in Section 4OA7 of this report.

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period operating at 100 percent power. On February 15, 2013, the plant commenced a Technical specification (TS) required shutdown to repair a leaking tube inside the 'A' Component Cooling Water (CCW) Heat Exchanger. After completing the necessary repairs, the reactor was brought back to critical on February 21, 2013. Power was returned to 100 percent on February 23, 2013. The plant remained at or near 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

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:

- 'B' control room heating ventilation and air conditioning (HVAC) system with 'A' system out of service for maintenance;
- 'A' and 'C' CCW trains with 'B' CCW pump out of service for maintenance; and
- 'C' Auxiliary Feedwater (AFW) train with 'B' turbine driven auxiliary feedwater pump 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), TS requirements, outstanding work orders, 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 corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

From February 25-28, 2013, the inspectors performed a complete system alignment inspection of the service water 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 work orders 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)

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 28: west engineered safeguards room;
- Fire Areas 5 & 7: diesel generator 1-1 and fuel oil day tank rooms;
- Fire Area 23: turbine building general area elevations 625'/612'/607'; and
- Fire Areas 2 & 3: cable spreading room and 1-D switchgear room.

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

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

a. Inspection Scope

The inspectors reviewed selected risk important plant design features and licensee procedures intended to protect the plant and its safety-related equipment from internal flooding events. The inspectors reviewed flood analyses and design documents, including the UFSAR, engineering calculations, and abnormal operating procedures to identify licensee commitments. The specific documents reviewed are listed in the Attachment to this report. In addition, the inspectors reviewed licensee drawings to identify areas and equipment that may be affected by internal flooding caused by the failure or misalignment of nearby sources of water, such as the fire suppression or the circulating water systems. The inspectors also reviewed the licensee's corrective action documents with respect to past flood-related items identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors walked down the following plant area to assess the adequacy of watertight doors and verify drains and sumps were clear of debris and were operable, and that the licensee complied with its commitments:

- West engineered safeguards room

Specific documents reviewed during this inspection are listed in the Attachment to this report. This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings were identified.

1R07 Heat Sink Performance (71111.07T)

.1 Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed operability determinations, completed surveillances, vendor manual information, calculations, test and inspection results associated with the 'B' CCW heat exchanger. This heat exchanger was chosen based on its risk significance in the licensee's probabilistic safety analysis, its important safety-related mitigating system support functions, its operating history and its relatively low margin.

For the 'B' CCW heat exchanger, the inspectors reviewed documentation associated with testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling to ensure the heat transfer capability of the heat exchanger was being maintained. This was accomplished by verifying (1) the test method used was consistent with accepted industry practices, or equivalent; (2) the test conditions were consistent with the selected methodology; (3) the test acceptance criteria were consistent with the design basis values; and (4) the results of heat exchanger performance testing were appropriately analyzed. The inspectors also reviewed test results to verify the licensee appropriately considered differences between testing conditions and design conditions. Additionally, inspectors assessed the frequency of testing, based on trending of test results, was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values and if test results considered test instrument inaccuracies and differences.

For the CCW heat exchangers, the inspectors reviewed the methods and results of heat exchanger thermal performance tests. The inspectors verified the methods used to inspect and clean the heat exchangers were consistent with as-found conditions identified, expected degradation trends and industry standards. The inspectors also verified the licensee's inspect and clean activities had established acceptance criteria consistent with industry standards, and whether the as-found results were recorded, evaluated, and appropriately dispositioned such that the as-left condition was acceptable. Inspectors identified concerns related to the evaluation and trending of thermal performance testing results. These concerns are documented in the section 1R07.1b of this report.

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

The inspectors verified the performance of the ultimate heat sinks and safety-related service water systems and their subcomponents such as piping, intake screens, pumps, valves, etc. by tests or other equivalent methods to ensure availability and accessibility to the in-plant cooling water systems.

The inspectors reviewed the licensee's performance testing of service water system and ultimate heat sinks. This included the reviewing the licensee's performance test results for key components and service water flow balance test results. In addition, the inspectors compared the flow balance results to system configuration and flow assumptions during design basis accident conditions. The inspectors also verified that the licensee ensured adequate isolation during design basis events, consistency between testing methodologies and design basis leakage rate assumptions, and proper performance of risk significant non-safety related functions.

The inspectors walked down the service water intake structure to verify the licensee's assessment on structural integrity and component functionality. This included verifying that the licensee ensured proper functioning of traveling screens and strainers, and structural integrity of component mounts. In addition, the inspectors verified the service water pump bay silt accumulation was monitored, trended, and maintained at an acceptable level by the licensee, and that water level instruments were functional and routinely monitored. The inspectors also verified the licensee's ability to ensure functionality during adverse weather conditions; that adequate water would still flow past sand-limiting underwater weir walls during periods of low lake level; and that the licensee had adequately protected against silt introduction during periods of low flow or low level.

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

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

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control", for failure to establish testing to demonstrate the safety-related CCW heat exchangers would perform satisfactorily in service.

Description: In a letter dated January 29, 1990, the licensee provided their response and program commitments to address Generic Letter (GL) 89-13, "Service Water System Problems Affecting Safety-Related Equipment." In this letter the licensee committed to test heat transfer capability of the A and B CCW heat exchangers. In addition, the licensee included actions under their GL 89-13 program to inspect, clean, and eddy current test the CCW heat exchangers in accordance with recommended Action 2 of GL 89-13.

The first test was performed in the 1990 refueling outage using Procedure T-300, "Determination of Heat Transfer Capability of CCW Heat Exchangers E-54A and E-54B." The results were documented in Attachment 4 of Procedure T-300 dated August 17, 1990, and AR A-NL-90-32 dated February 14, 1991. The licensee identified the B CCW heat exchanger was performing below design, but remained capable of removing required accident loads. The licensee determined additional performance testing was necessary in subsequent outages to perform trending and to ensure that significant degradation in heat transfer capability did not occur. The licensee implemented two

additional actions to perform cleaning of both heat exchangers in subsequent outages, and modified the proceduralized temperature collection method to correct instrument error issues that are created when using control room indication for these tests.

The second opportunity to implement testing on CCW heat exchangers was during the 1992 refueling outage. Heat transfer tests, again using procedure T-300, were performed. The results of these tests were evaluated in a licensee document titled "Review of SWS/CCW Testing," dated December 1, 1993. The review concluded the "T-300 test does not validate the design..." based on several factors including data scatter, instrument uncertainty, and CCW flow measurement.

In early 1994, the NRC performed a Service Water System Operational Performance Inspection (Inspection Report 05000255/94002 (Division of Reactor Safety) [ML9403090039]). In Attachment B of this report, the team considered the licensee's response adequate in addressing GL 89-13 concerns in aggregate because the licensee performed inspection, cleaning, and periodic maintenance each outage. However, the team also provided specific observations related to deficiencies in the thermal performance testing of the CCW heat exchangers, noting specifically "the performance testing portion of this program… exhibited a lack of preplanning and understanding of instrumentation accuracy's limitations and its impact on test results." The team concluded, "With the revision to improve CCW heat exchanger performance test reliability and with continued implementation of inspection, cleaning and periodic maintenance of safety-related heat exchangers, the GL [Recommended Action II] intent will be met."

The licensee implemented procedure T-300 two additional consecutive outages in 1995 and 1996, after enhancements were implemented, to attempt to correct deficiencies that contributed to the earlier thermal performance test flaws as identified in Deviation Report D-PAL-93-272 and the licensee's document dated December 1, 1993. The licensee showed, after data analysis, that both heat exchangers met the imposed as-found requirement for calculated overall heat exchanger fouling. As-left or post-cleaning fouling could not be determined because the system heat load needed to provide meaningful results was no longer present.

In 1998, the licensee began utilizing a new test methodology while it was in the demonstration phase of development, to verify the heat transfer capability of the CCW heat exchangers. The methodology was implemented through Procedure T-390, "Single Tube Testing of the CCW Heat Exchangers." Plant engineering and Vendor Support Engineering concluded, from a fouling perspective, the data was useable and indicated the pre-clean and post-clean condition of the 'B' CCW heat exchanger was maintained well below the calculated maximum fouling value, as documented in Electric Power Research Institute Technical Report TR 1009839. Based on these results, the licensee made the decision to only clean one CCW heat exchanger per outage, alternating so that each heat exchanger would be allowed to be cleaned every other outage. This frequency extension was supported by performance test data and ensured the limiting condition for operation fouling factors were not exceeded during the 3-year (2R) program frequency.

The licensee performed GOTHIC Analyses to ensure containment response acceptance criteria were met under the most limiting large break LOCA conditions, with one CCW

heat exchanger isolated, and assuming a fouling factor 2 times greater than design fouling factor in the operating heat exchanger.

The licensee implemented GL 89-13 program commitments, which included thermal performance testing, eddy current testing, inspection, cleaning, and maintenance on each CCW heat exchanger, at the alternating 2R frequency, from 1998 through 2009. The licensee discontinued the collection of post-clean fouling data after the third thermal performance test was performed on the A CCW heat exchanger in 2004.

On May 20, 2010, the licensee initiated Technical Justification 2010 to change the program frequency for performance testing the CCW heat exchangers from 2R to 3R (every 4.5 years). This change moved all subsequent inspection, cleaning, and testing for the A CCW and B CCW heat exchangers to 2012 and 2013 respectively. The licensee's justification was based on factors that included (1) heat exchanger manufacture, age, and service condition; (2) relatively clean lake water; (3) secondary side chemistry control; (4) minimizing time shut down cooling is reduced by 50 percent to allow testing and inspection of the heat exchangers; (5) historical thermal performance test results for the A CCW heat exchanger, and (6) historical eddy current test data for both heat exchangers.

Following the implementation of Technical Justification 2010, the licensee initiated another frequency reduction using Technical Justification 2012, dated March 27, 2012. This frequency change required that only 1 CCW heat exchanger be subject to program requirements every six years. Therefore, with acceptable results, each individual heat exchanger would be subject to inspection, cleaning, and performance testing approximately every 12 years. The inspectors had the following concerns:

- The UFSAR states, "In response to GL 89-13, a program was established to address the issue of bio-fouling of the service water system," and includes a reference to the licensee's initial response and program commitments documented in the letter dated January 29, 1990. Commitment 9 of this letter stated, the license "would test the heat transfer capability of the A and B CCW heat exchangers.

  NUREG-1871 "Safety Evaluation Report Related to The License Renewal of Palisades Nuclear Plant," states "aging effects are managed through (a) monitoring and control of bio-fouling, (b) flow balancing and flushing, (c) heat exchanger testing, and (d) routine inspection and maintenance program activities to ensure that aging effects do not impair component intended function. Inspection methods include visual, ultrasonic, radiographic, and eddy current tests. This program is responsive to NRC Generic Letter (GL) 89-13." GL 89-13, Supplement 1 states, "the final determined testing frequency should not be less than once every five years."

  Therefore, changing the frequency for inspection, cleaning, maintenance, performance testing, and eddy current testing of these heat exchangers to 12 years, does not meet the intent of the Generic Letter or program commitments.

- In Technical Justification 2012, the licensee created a table comparing the as-found and the as-left performance test data for the CCW heat exchangers. The

Single Tube Test Method determined the improvement that cleaning had on the heat exchanger after a single 2R period to ensure the fouling over the subsequent 2R period remained below design fouling factor limitations. This data was used to project future performance to determine the acceptability of extending the time between heat exchanger cleanings to 12 years. However, the licensee did not provide sufficient data to correlate this 2R period (3 years) to the potential expected fouling over a 12-year period.

  - For example, the licensee did not address the fact the as-found data represented a 3-year cleaning between tests and failed to consider the increase in accumulated fouling over individual and successive 2R periods. As such, the inspectors concluded this table could not be used as-is to demonstrate the CCW heat exchangers would be capable of performing their intended heat removal function during the extended period.

- The inspectors questioned the as-found performance test data collected between 1998 and 2007 that was used to support the frequency reduction. The inspectors noted the licensee did not incorporate B CCW heat exchanger data in their evaluation. In addition, the inspectors noted the 2003 data for the B CCW heat exchanger was not included in the table. Once located, the inspectors noted that the test showed an unexpected improvement in the fouling factor which had not been evaluated by the licensee at that time.

- Licensee administrative procedures SEP-HX-PLP-001 “Heat Exchanger Condition Assessment Program” and SEP-SW-PLP-001 “Raw Water Corrosion Program,” allow technical justifications to be performed when a need to deviate from these procedures is identified. However, instead of using a technical justification to deviate from these procedures, the licensee used the technical justifications to permanently change the performance test frequency located in the Master Heat Exchanger Testing Plan, which is provided as an attachment to SEP-HX-PLP-001. Revising the procedure would have required additional review and evaluation.

The licensee documented the inspectors’ concerns in CR-PLP-2013-00544 to address the program frequency concerns. The licensee implemented corrective actions to increase the frequency for implementing program requirements to ensure each CCW heat exchanger is reviewed at a minimum of once every 5 years.

It should be noted that on February 9, 2013, the licensee identified a lowering trend in CCW surge tank level; the licensee was able to stabilize tank level by adjusting the make-up control valve. Licensee operations staff requested the performance of an operability evaluation to determine operability with an unidentified leak. Compensatory measures were put in place to (1) ensure the CCW system would meet its 30 day mission time and (2) identify the leak using a tracer in the system. The licensee continued monitoring the trend until on February 14, 2013, indications of a thru-wall tube leak were identified on ‘A’ CCW heat exchanger. The licensee subsequently manually shut down the unit on February 15, 2013, to inspect and repair the failed heat exchanger tube.

On February 17, 2013, the licensee visually identified one leaking tube and one leaking tube plug. Follow up inspections and eddy current testing of approximately 20 percent

(398 out of 2021 tubes) of A CCW heat exchanger tubes revealed nine obstructed tubes and minor leakage/degradation of seven previously plugged tubes. All of these tubes were plugged during this maintenance activity.

The inspectors observed the licensee's follow-up activities to identify the source of the system leakage and the actions to identify the cause of the failure. The licensee initiated Root Cause Evaluation Report CR-PLP-2013-00652 to investigate the cause of the tube failure and to assess whether further changes were needed in the Heat Exchanger Monitoring Program.

Analysis: The inspectors determined the failure to establish testing to demonstrate that the safety-related CCW heat exchangers would perform satisfactorily in service is contrary to 10 CFR Part 50, Appendix B, Criterion XI, and is a performance deficiency. The performance deficiency was more than minor because it was associated with the equipment performance attribute of the Mitigating Systems Cornerstone and adversely affected the cornerstone objective of ensuring the availability reliability and capability of systems needed to respond to initiating events to prevent undesired consequences. Specifically, the inappropriate test frequency affected the licensee's ability to ensure the CCW heat exchangers were available and capable to reliably perform as expected.

The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," issued on June 19, 2012. Because the finding impacted the Mitigating Systems Cornerstone, the inspectors screened the finding through IMC 0609 Appendix A, "The Significance Determination Process for Findings At-Power," issued June 19, 2012, using Exhibit 2, "Mitigating Systems Screening Questions." The finding screened as of very low safety significance (Green) because the inspectors answered 'no' to all questions in Section A. Specifically, the inadequate test program was not a design deficiency and did not result in a loss of system or component function.

This finding has a cross-cutting aspect in the area of human performance, decision making component. The licensee did not use conservative decision making and did not conduct effectiveness reviews of safety significant decisions to verify the validity of underlying assumptions, identify possible unintended consequences, or determine how to improve future decisions. Specifically, the licensee failed to use conservative decision-making or verify the validity of underlying assumptions when evaluating the effect that reducing the frequency of testing, inspection, cleaning, and maintenance would have on the CCW heat exchangers. (H.1(b)).

Enforcement: 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," states, in part, "a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures, which incorporate the requirements and acceptance limits contained in applicable design documents." Procedures, EM-09-16 "Heat Exchanger Condition Assessment Program," Revision 9, and SEP-SW-PLP-001 "Raw Water Corrosion Program," Revision 0, state "the minimal final testing frequency for safety-related heat exchangers is once every 5 years." These procedures implement testing for safety related components. The renewed facility operating license section of the UFSAR, 1.9.1.14 states "the Open Cycle Cooling Water

Program is responsive to NRC GL 89-13," and GL 89-13 Recommended Action II states, “the minimum final test program frequency should be once every 5 years.”

Contrary to the above, since March 27, 2012, the licensee failed to establish testing to demonstrate that the safety-related CCW heat exchangers would perform satisfactorily in service. Specifically, the licensee failed to demonstrate the heat exchanger fouling factors would remain acceptable to ensure adequate heat transfer capability when the inspection, cleaning, eddy current testing, and thermal performance testing frequency was changed from every 4.5 years to 12 years. In addition, the licensee did not ensure the final test frequency was consistent with Generic Letter 89-13 recommendations, as committed to in licensee procedures EM-09-16 and SEP-SW-PLP-001 and UFSAR Section 1.9.1.14. Because this finding is of very low safety significance and has been entered into the licensee’s CAP as CR-PLP-2012-05132 and CR-PLP-2013-00544, this violation is being treated as a NCV consistent with the NRC Enforcement Policy (NCV 05000255/2013002-01, “Failure to Establish an Acceptable Component Cooling Water Heat Exchanger Final Test Frequency.”)

.2 Annual Heat Sink Performance (71111.07)

a. Inspection Scope

On February 15, 2013, the plant commenced a TSs required shutdown due to a tube leak in the ‘A’ CCW heat exchanger. The inspectors reviewed the licensee’s inspection and testing of the ‘A’ CCW heat exchanger during the forced outage in addition to the extent of condition and extent of cause assessments. Repair efforts were also observed and post-maintenance testing was reviewed. Repairs included plugging the affected tube, plugging blocked tubes, and re-plugging tubes noted to be leaking. A finding regarding implementation of an engineering change to re-plug tubes with a new design is discussed in Section 1R18 of this report. The inspectors (including support from a regional specialist) concluded that the inspections and repairs made to the heat exchanger were acceptable to allow the heat exchanger to be returned to service.

This annual heat sink performance inspection sample remains open pending further inspection of both CCW heat exchangers later in the year.

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 January 23, 2013, the inspectors observed a crew of licensed operators in the plant’s simulator during a licensed operator annual requalification operating test 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)

a. Inspection Scope

On February 15, 2013, and February 21, 2013, the inspectors observed operations staff conducting activities in the control room during a forced outage to shut down and start up the reactor. 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 and control rod manipulations; and
- oversight and direction from supervisors.

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

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

b. Findings

No findings were identified.

.3 Biennial Written and Annual Operating Test Results (71111.11A)

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the Biennial Written Examination, and the Annual Operating Test administered by the licensee from January 7, 2013 through February 15, 2013, as required by 10 CFR 55.59(a). The results were compared to the thresholds established in Inspection Manual Chapter 0609, Appendix I, "Licensed Operator Requalification Significance Determination Process (SDP)," to assess the overall adequacy of the licensee's Licensed Operator Requalification Training (LORT) program to meet the requirements of 10 CFR 55.59. (02.02)

This inspection constituted one annual licensed operator requalification examination results sample as defined in IP 71111.11-05.

a. Findings

No findings were identified.

.4 Biennial Review (71111.11B)

b. Inspection Scope

The following inspection activities were conducted during the week of January 14, 2013, to assess: 1) the effectiveness and adequacy of the facility licensee's implementation and maintenance of its systems approach to training (SAT) based LORT Program, put into effect to satisfy the requirements of 10 CFR 55.59; 2) conformance with the requirements of 10 CFR 55.46 for use of a plant referenced simulator to conduct operator licensing examinations and for satisfying experience requirements; and 3) conformance with the operator license conditions specified in 10 CFR 55.53. The documents reviewed are listed in the Attachment to this report.

- Licensee Requalification Examinations (10 CFR 55.59(c); Systems Approach To Training Element 4 as Defined in 10 CFR 55.4): The inspectors reviewed the licensee's program for development and administration of the LORT biennial written examination and annual operating tests to assess the licensee's ability to develop and administer examinations that are acceptable for meeting the requirements of 10 CFR 55.59(a).

  - The inspectors observed the administration of the annual operating test to assess the licensee's effectiveness in conducting the examination(s), including the conduct of evaluations of individual operator and crew performance, and post-examination analysis. The inspectors evaluated the performance of one crew in parallel with the facility evaluators one dynamic simulator scenario. (02.05)

- Conformance with Operator License Conditions (10 CFR 55.53): The inspectors reviewed the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53(e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators, and which control room positions were granted watch-standing credit for maintaining active operator licenses. (Medical records for

licensed operators were reviewed for compliance with 10 CFR 55.53(l) during the previous inspection in February 2012). (02.08)

This inspection constituted one biennial licensed operator regulation program sample as defined in IP 71111.11-05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- 480 volt breakers; and
- Instrument air.

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 two quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- online turbine vibration panel power supply replacement;
- elevated temperature on Z-Phase of main generator disconnect;
- online work on turbine stop valve;
- 'A' CCW heat exchanger Isolation to prepare system for work during forced outage;
- 1-2 emergency diesel generator planned maintenance outage; and
- impacts of removing the west engineered safeguards floor plug online.

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 six 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:

- oversized primary coolant pump impeller;
- emergency diesel generator ventilation system with one ventilation train out-of-service for maintenance;
- surveillance testing frequency restrictions on the emergency diesel generators to maintain containment spray pump operability; and
- CCW system leakage.

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 four samples as defined in IP 71111.15-05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following modification:

- tube plugging in 'A' component cooling water heat exchanger during forced outage.

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. 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 sample constitutes one permanent modification sample as defined in IP 71111.18-05.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of 10CFR50 Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to properly plan and document work on the safety-related 'A' CCW heat exchanger during a forced outage to repair leaks in the heat exchanger.

Work instructions for re-plugging several tubes did not contain critical acceptance criteria or documentation validating that an engineering change had been properly implemented.

Description: On February 15, 2013, the plant was shut down to repair tube leakage in the 'A' CCW heat exchanger. During the course of repair work the licensee decided to re-plug several other tubes in the heat exchanger, which were noted to have minor leakage through existing plugs. The old plugs used a seal-welded design. The licensee decided to re-plug the tubes with a mechanical-type seal plug and initiated the engineering change (EC) process to justify the change. Work was allowed to proceed on the heat exchanger while the EC was being processed. This is allowed per procedure EN-DC-115, Engineering Change Process. As work proceeded in the field and on the EC, information was obtained indicating certain parameters would be essential to ensure adequate integrity of the plugs. One was a requirement for roundness of the hole the plug would be inserted into; the other was the depth that the plug should be inserted to ensure sufficient interference fit when later hammered in. At this point, field work had proceeded to the point where the plugs requiring the EC were to be inserted. However, the acceptance criteria that evolved in processing the EC had not been incorporated into the work instructions. Omitting approved acceptance criteria was contrary to procedure EN-WM-105, Planning, which requires a level of detail in work instructions to be commensurate with the significance and complexity of the work. Examples are provided as to how this requirement can be met given the type of work package. The inspectors determined the level of detail was not sufficient and questioned the licensee. The licensee stopped and sought what changes would be necessary to the work instructions. However, the initial revision only included one of the critical parameters. After further questioning by inspectors, the other parameter was added. The next day, the inspectors noted that a condition report was not going to be written regarding the changes to the work instruction and questioned the licensee. A condition report subsequently written however, the changes were deemed enhancements with no further investigation into work management or engineering process issues. Besides the requirements in EN-WM-105, procedure EN-DC-115 requires controls to ensure that implementation of approved engineering changes have been properly verified. The completed work instructions did not contain documentation which would permit clear verification that the EC had been implemented correctly. Through interviews with workers and the fact that no leakage was observed after plug installation, the licensee determined the plugs had been installed per the approved EC. However, a corrective action was generated to formally document the results. During licensee review of the issue, it was also discovered that one method to validate leak-tightness had been marked as '"not-applicable" without a documented basis. Not documenting the basis was contrary to requirements in procedure EN-HU-106, Procedure and Work Instruction Use and Adherence. While the inspectors determined it was acceptable to not perform that method, not following the requirements to properly document the decision caused confusion with other workers and highlighted the programmatic issue observed with the control of some aspects of the heat exchanger work.

Analysis: The failure to adequately plan and document work associated with the safety-related 'A' CCW heat exchanger per procedures EN-DC-115 and EN-WM-105 was a performance deficiency warranting a significance determination. The issue was determined to be greater than minor per IMC 0612, Appendix B, "Issue Screening," issue date September 7, 2012, because if left uncorrected, it could lead to a more significant safety concern. The inspectors' decision was informed by examples 3j and 3k in

IMC 0612, Appendix E, "Examples of Minor Issues," issue date August 11, 2009. The examples refer to an issue not being minor if significant programmatic deficiencies were identified with the issue that could lead to worse errors if left uncorrected. When the issue was first raised by the inspectors, only one of the two critical parameters was initially added to the revised work instructions. Additionally, a condition report was not generated until further inspector questioning. When written, it referred to the missing criteria as enhancements and was closed, without a broader look at work management or engineering process issues. Further, two examples of inadequate documentation were identified. A basis for removing steps to check for leaks was not properly documented; and it was not clear from the completed work packages that the engineering acceptance criteria were met. Given these issues, the inspectors determined the threshold for a finding was met.

The inspectors concluded the finding adversely impacted the Mitigating Systems Cornerstone and was of very low safety significance (Green) utilizing IMC 0609, "Significance Determination Process," issue date June 2, 2011. Specifically, in Attachment 4, issue date June 19, 2012, utilizing Exhibit 2 of Appendix A, all questions in Section A were answered 'no,' since there was no loss of safety function to the component. The finding had an associated cross-cutting aspect in the work control component of the human performance area. Specifically, the licensee did not coordinate work activities by incorporating actions to ensure interdepartmental alignments were made while planning and executing the work to assure plant and human performance (H.3(b)).

Enforcement: 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished be included in work instructions. Work Order 265456 implements activities for work on a safety related component, the A CCW heat exchanger. Contrary to these requirements, on February 19, 2013, the inspectors identified that work instructions specified in Work Order 265456, Repair E-54A Heat Exchanger, to install mechanical plugs in the safety-related 'A' CCW heat exchanger did not include appropriate quantitative or qualitative acceptance criteria for determining that important activities had been satisfactorily accomplished. Specifically, tolerances to ensure the proper interference fit of plugs used to stop leakage and requirements to document the installation such that an approved engineering change could be validated were not present in the authorized work instructions being utilized in the field. The licensee interviewed workers to ensure plugs already installed were done correctly and revised the work instructions for plugs not yet installed. The licensee also entered the issue into the CAP as CR-PLP-2013-00773 and CR-PLP-2013-00969. This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's CAP. (NCV 05000255/2013002-02, Inadequate Work Instructions for Component Cooling Water Heat Exchanger.)

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:

- ‘B’ high pressure safety injection pump maintenance;
- battery charger #4 maintenance;
- ‘A’ CCW heat exchanger repairs;
- ‘A’ service water pump shaft sleeve replacement;
- reactor protective system bistable replacement; and
- 1-2 emergency diesel generator maintenance.

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

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

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Other Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled outage that began on February 15, 2013, and continued through February 22, 2013. The plant was shut down to repair a tube leak in the ‘A’ CCW heat exchanger. The affected tube was plugged, some tubes with potential blockage were plugged, and some previously plugged tubes with minor leaks were re-plugged. The licensee also replaced a degraded electrical disconnect in the switchyard that was being monitored prior to the outage.

The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule; observed portions of the reactor shutdown and cooldown; reviewed outage equipment configuration and risk management, electrical lineups, selected clearances; verified adequate control and monitoring of decay heat removal; verified adequate control of containment activities; observed portions of startup and heatup activities; and verified identification and resolution of problems associated with the outage were entered into the licensee’s CAP.

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

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:

- 'B' CCW pump testing (in-service test);
- 24 hour emergency diesel generator surveillance testing (routine);
- calibration of nuclear instruments (routine);
- safety injection testing (routine); and
- 'A' AFW system surveillance testing (routine).

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

- did preconditioning occur;
- the effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- acceptance criteria were 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 in-service testing activities, testing was performed in accordance with the applicable version of Section XI, American Society of Mechanical Engineers 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 four routine surveillance testing samples and one in-service testing sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

Introduction: A self-revealed finding of very low safety significance (Green) and an associated NCV of 10 CFR 50, Appendix B, Criterion V was identified for the failure to conduct the 'A' AFW pump technical specification surveillance test in accordance with the prescribed in-service test procedure. Specifically, the packing was adjusted when it was not required by the test procedure, which caused the pump packing to overheat and smoke. This necessitated emergent repair work and unplanned inoperability time for the pump.

Description: On March 15, 2013, operations and maintenance personnel were conducting the quarterly technical specification required surveillance test for the 'A' AFW pump per procedure QO-21. Step 4.3.10 of QO-21 states that pump packing leakage is "acceptable as long as there is some leakage with no steaming…[and] leakage that results in water spraying out of the packing is undesirable since prolonged operation could result in water entering the shaft bearing housing." Step 5.2.8 in the pump start section states to observe the pump packing and record "as-found" leakage, and then provides an "If/Then" table with guidance on adjusting leak-off, if needed. This table states: "1. if no leakage is observed or if steaming is observed, then the pump should be stopped and maintenance contacted to adjust the packing; 2. if leakage is spraying from the packing, then contact maintenance to adjust or have operations adjust it per the system operating procedure; or 3. if some leakage is observed with no steaming, then the packing leakage is acceptable."

From a review of the place-kept procedure after the event, the first two "if's" in the table were "N/A'd" and the third "if" was circle-slashed, indicating there was some leakage observed with no steaming and that the leakage was acceptable. From interviews after the event with the personnel involved, the licensee determined that based on skill-of-the-craft knowledge, a decision was made to adjust the packing anyway because the inboard packing gland area was hot to the touch. The packing was adjusted in ½ flat increments with 5 minute wait times in between increments. After three ½ flat adjustments the packing gland temperature appeared to stabilize, until a few minutes later when the field operator reported to the control room that there was smoke coming from the pump. The reactor operator in the control room immediately tripped the pump. An engineering review after the event identified that the packing adjustments moved the

packing in such a way that the leak-off flow was diverted outside of the packing (instead of inside where it could act as a coolant), causing the seal ring to overheat and glaze over, producing the smoking that was seen on the outside of the pump. The review also determined that an adjustment was not necessary in this circumstance based on the particular design of the pump's packing.

Analysis: The inspectors determined that the failure to conduct the 'A' AFW pump technical specification surveillance test in accordance with the prescribed in-service test procedure was a performance deficiency that warranted a significance determination. The inspectors determined that the issue was more than minor in accordance with IMC 0612 "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it affected the Mitigating Systems Cornerstone attribute of human performance, adversely impacting the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, a packing adjustment was made without being required by the procedure, causing the pump packing to overheat, which resulted in unplanned inoperability of the safety-related and risk significant 'A' AFW pump. Utilizing IMC 0609, "Significance Determination Process," issue date June 2, 2011, Appendix A, Exhibit 1, effective July 1, 2012, the finding screened as Green by answering "no" to the mitigating structures, systems and components, and functionality questions related to a loss of system or function or a loss of a train for greater than the technical specification limit.

The finding had an associated cross-cutting aspect in the area of human performance related to the cross-cutting component of resources, in that the licensee ensures plant personnel have complete, accurate, and up-to-date design documentation, procedures, and work packages. In this finding, the fact that the 'A' AFW pump has a unique packing design was not evident in the procedure being used and was not discussed during the pre-job briefs. Skill-of-the-craft knowledge was used to make the packing adjustments based on what the individual was qualified to do per the site's training procedure, but was not aware that this pump's packing design was unique. Additionally, the apparent cause evaluation for this event revealed that the test procedure in-use did not contain the same level of detail for making packing adjustments for this pump that the system operating procedure has, which would be used by operations personnel to make adjustments if maintenance was not available. The system operating procedure has specific values for the degree of flat adjustment and time periods to wait in between, which are different than how the adjustments were made during this event (H.2(c)).

Enforcement: 10 CFR Appendix B, Criterion V, requires, in part, that activities affecting quality shall be accomplished in accordance with instructions, procedures, or drawings appropriate to the circumstances. QO-21, is a quality procedure for testing the 'A' AFW pump. Contrary to this, on March 15, 2013, the licensee did not accomplish the technical specification surveillance test of the 'A' AFW pump as described in the in-service test procedure QO-21, Revision 40. The deviation from this procedure caused an unplanned inoperability of the pump to affect repairs. The licensee documented this issue in condition report CR-PLP-2013-01128, and performed an apparent cause evaluation. This violation is being treated as an NCV, consistent with section 2.3.2 of the Enforcement Policy because it was of very low safety significance and was entered into the licensee's CAP. Corrective actions included planned revisions to QO-21 to incorporate the guidance in the SOP for packing adjustments and to ensure this issue is discussed as operating experience in the pre-job briefing package for the surveillance

test. (NCV 05000255/2013002-03, Damage to 'A' AFW Pump Packing During Surveillance Run)

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on February 27, 2013, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator control room and the technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the Attachment to this report.

This emergency preparedness drill inspection constituted one sample as defined in IP 71114.06-05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstones: Public Radiation Safety and Occupational Radiation Safety

2RS4 Occupational Dose Assessment (71124.04)

This inspection constituted a partial sample as defined in IP 71124.04-05.

.1 External Dosimetry (02.02)

a. Inspection Scope

The inspectors evaluated whether the licensee's dosimetry vendor is National Voluntary Laboratory Accreditation Program (NVLAP) accredited and if the approved irradiation test categories for each type of personnel dosimeter used are consistent with the types and energies of the radiation present and the way the dosimeter is being used (e.g., to measure deep dose equivalent, shallow dose equivalent, or lens dose equivalent).

b. Findings

No findings were identified.

.2 Internal Dosimetry (02.03)

Routine Bioassay (In Vivo)

a. Inspection Scope

The inspectors reviewed procedures used to assess the dose from internally deposited nuclides using whole body counting equipment. The inspectors evaluated whether the procedures addressed methods for differentiating between internal and external contamination, the release of contaminated individuals, the route of intake and the assignment of dose.

The inspectors reviewed the whole body count process to determine if the frequency of measurements was consistent with the biological half-life of the nuclides available for intake.

The inspectors reviewed the licensee's evaluation for use of its portal radiation monitors as a passive monitoring system to determine if instrument minimum detectable activities were adequate to determine the potential for internally deposited radionuclides sufficient to prompt additional investigation.

The inspectors selected several whole body counts and evaluated whether the counting system used had sufficient counting time/low background to ensure appropriate sensitivity for the potential radionuclides of interest. The inspectors reviewed the radionuclide library used for the count system to determine its appropriateness. The inspectors evaluated whether any anomalous count peaks/nuclides indicated in each output spectra received appropriate disposition. The inspectors reviewed the licensee's 10 CFR Part 61 data analyses to determine whether the nuclide libraries included appropriate gamma-emitting nuclides. The inspectors evaluated how the licensee accounts for hard-to-detect nuclides in the dose assessment.

b. Findings

No findings were identified.

Internal Dose Assessment – Airborne Monitoring

a. Inspection Scope

The inspectors reviewed the licensee's program for airborne radioactivity assessment and dose assessment, as applicable, based on airborne monitoring and calculations of derived air concentration. The inspectors determined whether flow rates and collection times for air sampling equipment were adequate to allow lower limits of detection to be obtained. The inspectors also reviewed the adequacy of procedural guidance to assess internal dose if respiratory protection was used.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance (Green) and associated NCV of TS 5.4.1 for the failure to perform derived air concentration (DAC)-Hour tracking for airborne transuranic radioactivity as required by station

Procedure EN-RP-131, “Air Sampling” resulting in untimely internal dose assessments for selected plant workers.

Description: The inspectors identified an issue of concern in that the licensee did not perform effective DAC-Hour tracking and resultant dose assessments for airborne transuranic radioactivity for workers that were working in the reactor cavity and steam generator nozzle areas.

The NRC has established annual limit(s) on intake (ALI) for worker(s) that is the derived limit for the amount of radioactive material taken into the body of an adult worker by inhalation or ingestion in a year. The DAC means the concentration of a given radionuclide in air which, if breathed by the reference man for a working year of 2,000 hours under conditions of light work, results in an intake of one ALI. The term derived air concentration-hour (DAC-hour) is the product of the concentration of radioactive material in air and the time of exposure to that radionuclide(s), in hours. A licensee may take 2,000 DAC-hours to represent one ALI. The effective tracking of workers DAC-hour(s) exposures is a calculation method that is used to estimate workers’ committed effective dose equivalent (internal dose).

On April 16, 2012, the reactor cavity was posted as “Alpha Level 3” and “airborne radioactivity area.” This designation indicated that elevated concentrations of alpha emitting radionuclides were present in the reactor cavity. It also alerted the radiation protection staff of the presence of increased radiological hazards in the area, and that additional controls were required to effectively monitor and control workers’ internal radiation exposures. One of the additional requirements was to perform DAC-Hour tracking utilizing Procedure EN-RP-131, “Air Sampling,” Attachment 9.9, when the alpha air sample analysis results exceed 0.15 DAC. The actual air sample results indicated 0.21 DAC from alpha radionuclides and <0.1 DAC from beta-gamma radionuclides. The inspectors observed three work groups in the reactor cavity. One group was working on the tilt pit limit switch, one group was performing decontamination activities, and another group was preparing the reactor head to be raised. Some of the work groups were wearing respiratory protection while others were not.

When questioned by the inspectors, the licensee had difficulty supplying the requested Procedure EN-RP-131, “Air Sampling,” Attachment 9.9, DAC-Hour tracking form for some of the workers observed in the reactor cavity. The licensee indicated that there was a backlog of air samples that required more detailed radio-chemical analysis. This issue had been documented in the licensee’s CAP as CR-PLP-2012-02683. The corrective action document stated “Radiation protection airborne radioactivity air samples are not being analyzed in a timely manner.” However, effective corrective actions had not been implemented at the time of the initial NRC inspection. Delays in analyzing radioactive air sampling media can impact RP job coverage and worker’s dose assessment. The inspectors expanded the scope of review to include other “Alpha level 3” designated areas, including installation of the steam generator nozzle dams. Some workers installing the steam generator nozzle dams had their respiratory protection equipment fail while in-service. The licensee could not provide the inspectors the necessary DAC-Hour tracking information when requested. Procedure EN-RP-131, “Air Sampling,” required additional actions to be taken if the DAC Hour tracking form identified a worker that exceeded 4 DAC-Hours (10 millirem) in a 7-day period. This activity could not be completed for the workers performing steam generator nozzle dam installation and other work groups. This was an issue with work execution and

procedure adherence and not radiological work planning. Although alpha airborne monitoring and individual dose assessments were not performed in accordance with station procedures, there was sufficient oversight of the Airborne Radiation Program in place to identify significant changes in alpha radiation hazards at the plant. Specifically, the licensee was performing gross alpha monitoring of selected work activities.

Although the issue was initially documented in the licensee's CAP, the inspectors identified previously unknown weaknesses in the licensee's internal dose assessment process and in the corrective actions necessary to restore compliance. Consequently, the licensee reassessed their program for monitoring and controlling personnel in elevated alpha airborne areas of the plant. Additionally, the licensee performed internal dose assessments for workers that had entered and worked in all alpha airborne areas during the outage. This was a significant effort, in that, the licensee needed to evaluate each individual radiologically controlled area entry to determine which personnel had entered alpha airborne areas; determine the airborne isotopic mixes and isotopic concentrations; determine alpha airborne area exposure times; and then perform individual dose assessments. The doses were assessed based on station documentation of access to the airborne radioactivity areas through the radiological work control process and the monitoring and evaluation of airborne radioactivity samples collected during the outage. When uncertainties were identified, the licensee used conservative assumptions in order to determine bounding conditions. There were 59 workers that were assigned internal radiation exposure as a result of this issue with the highest individual exposure estimate of 56 mrem.

Analysis: The inspectors determined that the issue of concern was a performance deficiency because the licensee did not perform DAC-Hour tracking for airborne transuranic radioactivity as required by plant procedure EN-RP-131, "Air Sampling," resulting in untimely internal dose assessments for various plant workers. The inspectors determined that the cause of the performance deficiency was reasonably within the licensee's ability to foresee and correct and should have been prevented.

The issue was not subject to traditional enforcement since the incidents did not have a significant safety consequence, did not impact the NRC's ability to perform its regulatory function, and was not willful.

The performance deficiency was determined to be of more than minor safety significance in accordance with IMC 0612 Appendix B, "Issue Screening," issued September 7, 2012, because it was associated with the program and process attribute of the Occupational Radiation Safety Cornerstone and adversely affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation. Specifically, not performing DAC-Hour tracking for airborne transuranic radioactivity affected the licensee's ability to assess workers internal exposures in a timely manner, and adversely impacted the licensee's ability to monitor, control and limit radiation exposures (committed effective dose equivalent or internal dose). The inspectors also reviewed the guidance in IMC 0612 Appendix E, "Examples of minor issues," and did not find any similar examples.

In accordance with IMC 0609 Appendix C, "Occupational Radiation Safety Significance Determination Process," issue date August 19, 2008, the inspectors determined that the finding had very low safety significance (Green) because the finding: (1) did not involve ALARA planning and controls; (2) did not involve a radiological overexposure; (3) there

was not a substantial potential for an overexposure; and (4) there was no compromised ability to assess dose.

The inspectors determined that the primary cause of this finding was related to the use of site specific work instructions and thus related to the cross-cutting aspect of human performance in the component of resources. Specifically, that the licensee did not maintain complete, accurate and up-to-date procedures and work packages. H.2(c)

Enforcement: Technical Specification 5.4.1 states, in part, written procedures shall be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Section 7.e.3, recommends procedures for airborne radioactivity monitoring. Contrary to the above, as of April 19, 2012, licensee procedure EN-RP-131, "Air Sampling," was not implemented to perform DAC Hour tracking for airborne transuranic radioactivity. Corrective actions included re-evaluating the use of site specific work instructions and completing the required personnel dose assessments. Actual individual dose assessments were completed in November 2012. Since this finding and violation were of very low-safety significance and have been entered in the licensee's CAP as CR-PLP-2012-02683 this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-04; Failure to Perform DAC-Hour Tracking).

Internal Dose Assessment – Whole Body Count Analyses

a. Inspection Scope

The inspectors reviewed several dose assessments performed by the licensee using the results of whole body count analyses. The inspectors determined whether affected personnel were properly monitored with calibrated equipment and that internal exposures were assessed consistent with the licensee's procedures.

b. Findings

No findings were identified.

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08)

This inspection constituted one complete sample as defined in IP 71124.08-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the solid radioactive waste system description in the Final Safety Analysis Report, the Process Control Program, and the recent radiological effluent release report for information on the types, amounts, and processing of radioactive waste disposed.

The inspectors reviewed the scope of any quality assurance audits in this area since the last inspection to gain insights into the licensee's performance and inform the "smart sampling" inspection planning.

b. Findings

No findings were identified.

.2 Radioactive Material Storage (02.02)

a. Inspection Scope

The inspectors selected areas where containers of radioactive waste are stored, and evaluated whether the containers were labeled in accordance with 10 CFR 20.1904, “Labeling Containers,” or controlled in accordance with 10 CFR 20.1905, “Exemptions to Labeling Requirements,” as appropriate.

The inspectors assessed whether the radioactive material storage areas were controlled and posted in accordance with the requirements of 10 CFR Part 20, “Standards for Protection against Radiation.” For materials stored or used in the controlled or unrestricted areas, the inspectors evaluated whether they were secured against unauthorized removal and controlled in accordance with 10 CFR 20.1801, “Security of Stored Material,” and 10 CFR 20.1802, “Control of Material Not in Storage,” as appropriate.

The inspectors evaluated whether the licensee established a process for monitoring the impact of long term storage (e.g., buildup of any gases produced by waste decomposition, chemical reactions, container deformation, loss of container integrity, or re-release of free-flowing water) that was sufficient to identify potential unmonitored, unplanned releases or nonconformance with waste disposal requirements.

The inspectors selected containers of stored radioactive material, and assessed for signs of swelling, leakage, and deformation.

b. Findings

No findings were identified.

.3 Radioactive Waste System Walkdown (02.03)

a. Inspection Scope

The inspectors walked down accessible portions of select radioactive waste processing systems to assess whether the current system configuration and operation agreed with the descriptions in the Final Safety Analysis Report, Offsite Dose Calculation Manual, and Process Control Program.

The inspectors reviewed administrative and/or physical controls (i.e., drainage and isolation of the system from other systems) to assess whether the equipment, which is not in service or abandoned in place, would not contribute to an unmonitored release path and/or affect operating systems or be a source of unnecessary personnel exposure. The inspectors assessed whether the licensee reviewed the safety significance of systems and equipment abandoned in place in accordance with 10 CFR 50.59, “Changes, Tests, and Experiments.”

The inspectors reviewed the adequacy of changes made to the radioactive waste processing systems since the last inspection. The inspectors evaluated whether changes from what is described in the Final Safety Analysis Report were reviewed and documented in accordance with 10 CFR 50.59, as appropriate and to assess the impact on radiation doses to members of the public.

The inspectors selected processes for transferring radioactive waste resin and/or sludge discharges into shipping/disposal containers and assessed whether the waste stream mixing, sampling procedures, and methodology for waste concentration averaging were consistent with the Process Control Program, and provided representative samples of the waste product for the purposes of waste classification as described in 10 CFR 61.55, "Waste Classification."

For those systems that provide tank recirculation, the inspectors evaluated whether the tank recirculation procedures provided sufficient mixing.

The inspectors assessed whether the licensee's Process Control Program correctly described the current methods and procedures for dewatering and waste stabilization (e.g., removal of freestanding liquid).

b. Findings

No findings were identified.

.4 Waste Characterization and Classification (02.04)

a. Inspection Scope

The inspectors selected the following radioactive waste streams for review:

- dry active waste for 2011 and 2012;
- anion, cation, and mixed bed resins; and
- alternative radwaste processing system (ALPS) charcoal and resin beds.

For the waste streams listed above, the inspectors assessed whether the licensee's radiochemical sample analysis results (i.e., "10 CFR Part 61" analysis) were sufficient to support radioactive waste characterization as required by 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste." The inspectors evaluated whether the licensee's use of scaling factors and calculations to account for difficult-to-measure radionuclides was technically sound and based on current 10 CFR Part 61 analyses for the selected radioactive waste streams.

The inspectors evaluated whether changes to plant operational parameters were taken into account to: (1) maintain the validity of the waste stream composition data between the annual or biennial sample analysis update; and (2) assure that waste shipments continued to meet the requirements of 10 CFR Part 61 for the waste streams selected above.

The inspectors evaluated whether the licensee had established and maintained an adequate Quality Assurance Program to ensure compliance with the waste classification and characterization requirements of 10 CFR 61.55 and 10 CFR 61.56, "Waste Characteristics."

b. Findings

No findings were identified.

.5 Shipment Preparation (02.05)

a. Inspection Scope

The inspectors observed shipment packaging, surveying, labeling, marking, placarding, vehicle checks, emergency instructions, disposal manifest, shipping papers provided to the driver, and licensee verification of shipment readiness. The inspectors assessed whether the requirements of applicable transport cask certificate of compliance had been met. The inspectors evaluated whether the receiving licensee was authorized to receive the shipment packages. The inspectors evaluated whether the licensee's procedures for cask loading and closure procedures were consistent with the vendor's current approved procedures.

Due to limited opportunities for direct observation, the inspectors reviewed the technical instructions presented to workers during routine training. The inspectors assessed whether the licensee's Training Program provided training to personnel responsible for the conduct of radioactive waste processing and radioactive material shipment preparation activities.

b. Findings

No findings were identified.

.6 Shipping Records (02.06)

a. Inspection Scope

The inspectors evaluated whether the shipping documents indicated the proper shipper name; emergency response information and a 24-hour contact telephone number; accurate curie content and volume of material; and appropriate waste classification, transport index, and UN number for the following radioactive shipments:

- PLP-2011-RM-024;
- PLP-2011-RW-010
- PLP-2011-RW-011;
- PLP-2012-RW-018; and
- PLP-2012-RM-058.

Additionally, the inspectors assessed whether the shipment placarding was consistent with the information in the shipping documentation.

b. Findings

No findings were identified.

.7 Identification and Resolution of Problems (02.07)

a. Inspection Scope

The inspectors assessed whether problems associated with radioactive waste processing, handling, storage, and transportation, were being identified by the licensee at an appropriate threshold, were properly characterized, and were properly addressed for resolution in the licensee CAP. Additionally, the inspectors evaluated whether the corrective actions were appropriate for a selected sample of problems documented by the licensee that involve radioactive waste processing, handling, storage, and transportation.

The inspectors reviewed results of selected audits performed since the last inspection of this program and evaluated the adequacy of the licensee's corrective actions for issues identified during those audits.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams with Complications (IE04) performance indicator (PI) for the period from January 1, 2012, through December 31, 2012. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) 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 Integrated Inspection Reports for the period of January 1 2012, through December 31, 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 PI 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 with complications sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Heat Removal System (MS08) PI for the period from January 1, 2012, through December 31, 2012. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI 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, MSPI derivation reports, and NRC Integrated Inspection Reports for the period of January 1, 2012, through December 31, 2012, to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

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

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 (Closed) Unresolved Item 05000255/2012005-03, Safety Injection Refueling Water Tank Evaluation of Corrosion (This inspection is part of the additional inspections referenced in the Palisades Deviation letter.)

On March 25, 2011, the Palisades Nuclear Plant entered the period of extended operation of the renewed license. On May 18, 2011, leakage from the catacombs area beneath the Safety Injection Refueling Water (SIRW) tank was identified and documented in CR-PLP-2011-02491, "SIRW Tank Leakage." Ultimately the licensee determined the source of the leak was from the tank. Repairs to the tank were attempted during Refueling Outage 1R22; however, on June 12, 2012, the plant was shut down due to leakage from the SIRW tank exceeding operational decision-making issue process trigger point of 31 gallons per day.

During the review of CR-PLP-2012-4451, the inspectors became concerned the associated aging effects of the accumulated water were not properly managed. The inspectors were concerned because the accumulated water beneath the floor of the SIRW tank created an environment that could promote corrosion of the tank floor and nozzles and it did not appear the licensee formally evaluated this condition. The inspectors required more information to determine whether this issue constituted a finding of significance; therefore, the issue was considered unresolved.

Based upon further review during this inspection period, the inspectors noted the licensee conducted magnetic and ultrasonic inspection of the tank floor. The licensee concluded there was no measurable degradation. The licensee did identify some degradation of the F-nozzle in the form of dimpling caused by concrete-aluminum interaction. This degradation was evaluated in EC 38547 and PLP-RPT-12-00108 and

found to be acceptable for continued operation. The licensee implemented a periodic tank inspection task to manage the identified degradation. The inspectors found this to be acceptable. Documents reviewed as part of this inspection are listed in the attachment. This Unresolved Item (URI) is closed.

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

.4 Selected Issue Follow-up Inspection: Through Wall Leakage of Control Rod Drive Mechanism-24 (This inspection is part of the additional inspections referenced in the Palisades Deviation letter.)

a. Inspection Scope

On August 12, 2012, the licensee shut down the plant to investigate an increase in unidentified leakage. The leakage was determined to be from a crack in Control Rod Drive Mechanism (CRDM)-24. The NRC dispatched a special inspection team (SIT) to review the CRDM-24 leakage event. The results of that inspection were provided in Inspection Report 05000255/2012012. The licensee completed an evaluation to determine the cause of the cracking (CR-PLP-2012-05623).

From March 4, 2013, to March 15, 2013, the inspectors performed an in-depth problem identification and resolution sample based upon review of the licensee's root cause report (RCR) contained in corrective action document CR-PLP-2012-05623. In addition, the inspectors performed reviews related to three URIs identified during the SIT inspection:

- URI 05000255/2012012-01; TS for PCS Pressure Boundary Leakage. (The closure of this URI is documented in Section 4OA2.4 (b.1) of this report.);
- URI 05000255/2012012-02; Potential Inadequate Degradation Evaluation of CRDM Housings. (The closure of this URI is documented in Section 4OA5.2 of this report.); and
- URI 05000255/2012012-03; Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality. (The closure of this URI is documented in Section 4OA2.4 (b.1) of this report.).

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

b. Findings

1. Failure to Take Corrective Actions to Prevent Recurrence of Control Rod Drive Mechanism Housing Cracking and Leakage

Introduction: A self-revealing finding of very low safety significance with an associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, and TS 3.4.13 Primary Coolant System (PCS) Operational Leakage, was identified for failure to take corrective actions to prevent recurrence of CRDM cracking and leakage, a significant condition adverse to quality (SCAQ). As a result the plant was operated with PCS pressure boundary leakage. Specifically, for Criterion XVI, the licensee failed to include the internal CRDM housing weld build-up area within the scope of corrective actions taken for a 2001 CRDM through wall leak on CRDM-21 caused by transgranular stress corrosion cracking (TGSCC). Subsequently, a through wall leak recurred in the weld build-up area on CRDM-24 in 2012 due to TGSCC. As a result, the licensee operated with PCS pressure boundary leakage, which is not allowed by TS 3.4.13. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

Description: In 2001, the licensee discovered a steam leak in the housing of CRDM-21 caused by a through-wall TGSCC at CRDM housing weld No. 3, which was located just below the weld build-up region (weld No. 5). Weld No. 5 consists of a weld material deposit applied to the inside diameter (ID) of the CRDM housing, which provides for alignment of the CRDM. This issue was categorized as a SCAQ by the licensee (CPAL0102186) because it represented a break in the reactor system pressure boundary. The licensee's root cause evaluation was documented in RCR/C-PAL-01-02186 and concluded that the cracks in CRDM-21 were caused by TGSCC, which occurred in areas of heavy grinding or machining tool marks. Specifically, this leak was the result of an ID initiated, axially oriented, transgranular crack in the austenitic stainless steel housing material. The failure analysis performed in response to this event identified both axial and circumferential cracks associated with weld No. 3. Extent of condition inspections revealed additional, non-through wall cracks associated with weld No. 3 in 41 of the 44 remaining housings for a total of 42 of 45 housings containing cracks.

In response to the 2001 cracking, Palisades replaced all 45 CRDM housings with housings thought to be more resistant to cracking. Principle changes included:

- elimination of weld No. 2;
- relocation of weld No. 3 to a higher location thereby minimizing the deposition of crud in the gap between the weld and the bottom plate of the rack and pinion assembly;
- reduction in residual stresses and cold work on welds by requiring better surface finishes; and
- use of heat sink welding to reduce ID residual tensile stresses.

In January of 2002, an NRC SIT (reference IR 50-2555/01-15) reviewed the licensee proposed corrective actions associated with the through-wall leakage of the CRDM-21 housing caused by TGSCC. The 2001 RCR reviewed by the NRC stated the action to prevent recurrence was to "develop and implement an inspection plan to address areas and components identified in Attachment C-Extent of Condition." One of the

components included in Attachment C was the CRDM. The recommended action was to perform volumetric inspection of the welds contained in the CRDM. Subsequently, the licensee decided to change this action and exclude weld No. 5.

Following the subsequent 2012 CRDM-24 leak, the licensee determined the leak occurred because of a through-wall crack adjacent to weld No. 5. The licensee formed a root cause team (RCT) staffed with licensee personnel and augmented with input from vendors. The root cause investigation was conducted in accordance with site procedure EN-LI-118, "Root Cause Evaluation Process" and was documented in root cause analysis report CR-PLP-2012-05623. In this report, the licensee's RCT determined that the probable cause of the cracking was:

> "Stresses in the weld build up area due to manufacturing irregularities and misalignments between CRDM-24 upper housing, support tube, and the associated reactor head penetration/CRDM nozzle. Based on lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress."

The RCT also identified the following contributing cause:

> "TGSCC initiating within the internal weld build-up material of CRDM-24. The through wall crack initiated in the weld material and then propagated through the base metal until a leak developed in the outer diameter (OD) witness band region at the base of the ID weld build up."

This conclusion was based upon destructive and non destructive examinations (NDE) completed on a section of the failed housing, which included the through-wall flaw. The RCT also relied upon vendor technical reports assessing the results of the NDE as well as vendor calculations related to the stresses in the CRDM housings.

To determine the extent of condition, the licensee performed ultrasonic (UT) examinations of weld No. 5 on eight additional CRDM housings. The licensee selected these housings based on being in a similar location on the head as CRDM-24, and previous cracking having been identified in some of these housings prior to the replacement of the CRDM upper housings and seal housings in 2002. The inspectors concluded that this was an adequate sample for an initial extent of condition review based upon the concept that, in light of eight negative exams, the statistical probability of a flaw in the remaining CRDM housings was very low. Additionally, the licensee planned to conduct examinations of more housings during the next refueling outage.

The inspectors concluded that the licensee actions following the 2001 leak were not adequate because the appropriate actions to preclude recurrence were within the licensee's ability to foresee and implement. Specifically, the inspectors concluded that the licensee did not effectively implement corrective actions for the 2001 CRDM housing leak resulting in the 2012 CRDM-24 housing leak.

Licensee corrective actions taken in response to the 2001 event were limited to butt welds. The inspectors reviewed the licensee actions to determine if they had been sufficient to eliminate one of the three necessary factors to cause TGSCC on the CRDM housings: (1) a susceptible material, (2) a corrosive environment and (3) tensile stress. The inspectors identified that the licensee had failed to eliminate one or more of the

necessary factors at weld No. 5 (which was not a butt weld) to preclude TGSCC in the replacement housing.

Specifically:

- the licensee's 2001 RCR documented that weld No. 5 is exposed to essentially the same environment as the weld that experienced the cracking (corrosive environment remained unchanged);
- no analysis was completed on the stress conditions for weld No. 5 prior to approving the modified replacement housing design (the potential for residual tensile weld stresses on ID of CRDM surface was not ruled out by analysis and therefore, should have been considered);
- fabrication restrictions to prohibit grinding were not applied to weld No. 5 (grinding promotes residual tensile stress state on ID of CRDM surface);
- machining was performed on weld No. 5 during the fabrication process in order to achieve the dimensions and geometry specified in the design, which induced cold work stresses in the weld; and
- material was changed from Type 347 to Type 316 stainless steel (both materials are essentially equally susceptible to TGSCC).

Also, in 1991, the Fort Calhoun plant had experienced through-wall leakage due to TGSCC at weld No. 5 of its CRDM housings (same housing design) and this operational experience had been reviewed by the licensee and dismissed. In the licensee's 2001 root cause evaluation, the licensee reviewed the weld build-up region failure by TGSCC at Fort Calhoun and concluded it would not occur at Palisades. This conclusion was based on the assumption that a higher oxygen environment (more aggressive environment) would exist in the Fort Calhoun housings than in the inservice Palisades housings. However the licensee did not confirm this assumption, nor did the licensee perform additional testing to determine if the environment of its inservice housings was sufficiently benign to prevent TGSCC. The licensee's 2012 RCT documented that due to organizational/programmatic weakness at Palisades, the 1991 Fort Calhoun operating experience was not adequately utilized to include inspection of the weld No. 5. Similarly, the inspectors identified that the licensee had missed a key opportunity to implement effective corrective actions that could have prevented recurrence of the 2001 leakage event and had elected not to pursue that aspect further. Specifically, in EA-EAR-2001-0426-01, the licensee considered fabricating the replacement housings with Inconel 600 material because it was much more resistant to TGSCC, but ultimately decided not to do so. Additionally, various vendor reports were generated related to this issue in the mid 2000's. Those reports documented the potential susceptibility of weld No. 5 to TGSCC based upon a review of the CRDM housing conditions and available operating experience. The reports also noted that weld No. 5 was not inspected in any of the housings in 2001. One report in 2003 noted that weld No. 5 should have been examined as part of the action from the 2001 events since it was similar to Fort Calhoun. The issuance of these documents represented another opportunity for the licensee to identify the susceptibility of weld No. 5 to TGSCC prior to the cracking in CRDM-24.

The inspectors concluded the corrective actions taken in response to the 2001 CRDM through wall leak from TGSCC, a SCAQ, were not effective to preclude repetition. In particular, a through wall leak did recur on a CRDM from TGSCC. This issue was within the licensee's ability to foresee and correct; therefore, the issue was a performance deficiency. During the 2012 NRC special inspection, the NRC identified an URI for the

TS pressure boundary leak. TS Limiting Condition for Operations (LCO) 3.4.13 does not allow any PCS pressure boundary leakage. In particular, TS Basis B3.4.13 "PCS Operational Leakage," explains that "No pressure boundary leakage from within the primary coolant pressure boundary is allowed, being indicative of material degradation. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Not complying with this LCO could result in continued degradation of the primary coolant pressure boundary." Further, Action B, associated with this LCO, requires shutdown to Mode 3 in 6 hours and Mode 5 in 36 hours for such leakage. The licensee determined the CRDM-24 leakage commenced on or around July 14, 2012, and the plant continued to operate in this condition until August 12, 2012. Because the licensee was not aware of the existence of pressure boundary leakage, it failed to shut down the unit in six hours for a pressure boundary leak as required by TS 3.4.13 Action B. The NRC previously assessed the site's action for increasing unidentified leakage as part of the SIT. The NRC determined, at the time of higher unidentified leakage, the site took appropriate actions to attempt to locate the leak, eventually shutting down around .3 gallons per minute (gpm) leakage (earlier than the TS value of 1 gpm value for unidentified leakage). The licensee did not identify the source of the leakage as pressure boundary leakage until the shutdown on August 12, 2012, when a tour near the vessel head revealed the leaking housing. The pressure boundary leakage resulted in a TS violation and was due to the performance deficiency associated with the above mentioned Criterion XVI violation.

Based on the review discussed above, URIs 05000255/2012012-01 "TS for PCS Pressure Boundary Leakage" and 05000255/2012012-03 "Potential Failure to Take Corrective Actions to Prevent Recurrence of a Significant Condition Adverse to Quality" are closed.

Analysis: The inspectors determined that the licensee's failure to prevent recurrence of TGSCC of the CRDM housings (an SCAQ) that resulted in a violation of TS was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it adversely affected the Initiating Events Cornerstone objective to limit the likelihood of events that upset plant stability. The issue was associated with the attribute of equipment performance. Specifically, the licensee did not take adequate corrective actions to prevent recurrence of leakage in CRDM housings, which represents pressure boundary leakage. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 0609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors determined that the Initiating Events Cornerstone was affected because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1 "Initiating Events Screening Questions," in IMC 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the slow rate of change for leakage for this cracking mechanism and this type of material. Type 316 stainless steel

material under TGSCC will experience leakage rates well below a small break LOCA, which would be observed through the crack, alerting operators to take action to shut down the plant prior to experiencing a component rupture.

The cause of this finding, non-conservative decision making, occurred over ten years ago and is well outside of the nominal 3 year period in IMC 0612; and was not indicative of current performance, because no other opportunities to identify the issue occurred during the previous 3 year period. However more recently, the licensee exhibited non-conservative decision making with respect to addressing the potential for CRDM housing cracking and leakage during the recent root cause (Section 4OA2.4 (b.2) of this report), resulting in another finding. This cross-cutting aspect will be captured through the other finding.

Enforcement: The inspectors identified the following two violations of NRC requirements were associated with this finding:

- Title 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requires, in part, that, for significant conditions adverse to quality, the cause of the condition is determined and corrective action taken to preclude repetition.

  Contrary to the above, as of August 12, 2012, the licensee had failed to take corrective actions to preclude repetition for a SCAQ. Specifically, on June 21, 2001, the licensee discovered a through wall leak in CRDM-21 due to TGSCC and failed to reasonably include weld No. 5 in the corrective actions, which resulted in a subsequent through wall leak in CRDM-24 due to TGSCC.

- TS LCO 3.4.13 requires PCS operational leakage be limited to "No pressure boundary LEAKAGE" when in Modes 1 through 4.

  Contrary to the above, on or around July 14, 2012, PCS pressure boundary leakage at CRDM-24 existed while in Mode 1. Further, because the licensee was not aware that the leakage was PCS pressure boundary leakage, the licensee did not implement the associated TS action statement.

As a result of the second through- wall leak, the licensee took corrective actions, which included the development of an inspection plan that would inspect a sample of CRDM weld No. 5 every outage until all CRDM housings were inspected.

Because these violations were of very low safety significance and were entered into the licensee's CAP as CR-PLP-2013-01134, these violations are being treated as an NCVs, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2013002-05; Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage).

2. Failure to Adequately Address the Generic Implications of the Cracking identified in Control Rod Drive Mechanism-24

   Introduction: The inspectors identified a finding of very low safety significance (Green) with an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, for the licensee's failure to accomplish quality activities in accordance with the prescribed procedures. Specifically, the licensee failed to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with

procedure EN-LI-118, "Root Cause Evaluation." This issue was entered into the licensee's CAP under CR-PLP-2013-05623.

Description: As a result of the cracking identified in CRDM-24, which was characterized as a SCAQ, the licensee performed a root cause evaluation in accordance with procedure EN-LI-118. This procedure was identified as quality related and served to implement control pursuant to the licensee's Quality Assurance Program. While reviewing the 2012 RCR (CR-PLP-2013-05623) related to the cracking identified in CRDM-24, the inspectors identified that the licensee had not appropriately considered the generic implications of the cracking in the extent of condition review. The licensee's proposed corrective actions, as a result of the 2012 RCR, narrowly focused on weld No. 5. Instead of also including broader actions to ensure other CRDM housing welds were fit for their intended service life, the licensee's corrective actions consisted only of performing inspections of weld No. 5 on all CRDM housings.

On March 13, 2013, the inspectors requested that the licensee provide the bases for excluding other CRDM housing welds (weld No. 3 below weld No. 5 and weld No. 4 above weld No. 5) from the 2012 RCR scope of planned corrective actions. On March 29, 2013, the licensee provided additional information to justify excluding these welds from the scope of the corrective actions. The licensee credited the corrective actions associated with the modifications to the CRDM housing design completed in 2001 as the basis to exclude housing welds No. 3 and No. 4 from additional actions to identify the extent of TGSCC. The corrective actions taken in 2001 included performing heat sink welding, which is a methodology used to reduce the stresses on the inner ID of the weld. The licensee also changed the design to reduce design stresses at weld No. 3 and specified a smoother surface finish (RMS 125) to reduce potential crack initiation points. The licensee stated that these actions would produce compressive stresses on the ID of welds No. 3 and No. 4 making them immune from cracking. The inspectors acknowledged that these actions would reduce the tensile stress at the ID surface and thus reduce the probability of initiating TGSCC. However, the information provided did not demonstrate that TGSCC would not occur because it did not demonstrate that tensile stress would be eliminated at the ID surface during operation.

The inspectors identified that the three factors required for TGSCC could still be present at welds No. 3 and No. 4 as follows:

- Corrosive environment – Weld No. 3 would operate in a similar environment as weld No. 5 of the CRDM housing. Weld No. 4 would be exposed to a lower operating temperature than weld No. 5, however, TGSCC can still occur at 250 degrees Fahrenheit as evidenced by the Palisades previous operating experience with cracking identified in the seal housings that operate at even lower temperatures;
- Susceptible material – Welds No. 3 and No. 4 are composed of the same weld filler and base metal materials as weld No. 5 (e.g., weld filler material consistent with the Type 316 stainless housing base metal). This material would be equally susceptible to TGSCC, as the Type 347 stainless steel and weld filler materials used in the pre-2001 CRDM housing design that developed a through-wall leak caused by TGSCC at weld No.3; and
- Tensile stresses – While it is assumed that the corrective actions taken in response to the 2001 leak will reduce the potential for tensile stresses to exist on the inner surface of CRDM housings at welds No. 3 and No. 4, especially in light

of repairs made to welds No. 3 and No. 4, it had not been conclusively demonstrated that these tensile stresses have been eliminated. As such, when evaluating welds No. 3 and No. 4 for applicability to the 2012 root cause, it was not reasonable to conclude that tensile stresses were not present and that therefore, the potential for TGSCC had been eliminated.

The 2012 RCR discussed manufacturing irregularities and misalignment between CRDM-24 and the support tube, seismic supports, and the associated reactor head penetration/CRDM nozzle as potential source of stresses leading to cracking. However, the RCR also stated that "based on the lack of cracking found in the other eight upper housings tested, the failed CRDM-24 upper housing contains an as-yet unidentified additional stress." Because the cause of the additional stress was not identified, the licensee had not established a basis in the RCR to exclude welds No. 3 and No. 4 from the extent of condition review (e.g., potential generic implications). In 2001, assumptions on crack growth rate and inspection intervals for welds No. 3 and No. 4 were made based on the information known at the time. The 2001 crack went through-wall after the CRDM was in service for 30 years and the cracking was widespread among the other CRDM housings. In 2012, the crack propagated through-wall after the CRDM was in service for 11 years and the cracking did not appear as widespread. Though TGSCC was a factor in both cracking events, there are still unknowns associated with the 2012 incident. The unknown additional stresses, as well as the time the CRDM was inservice before cracking in 2012, represent key differences as related to the cracking identified in 2001. In the 2012 RCR, the licensee did not consider these or other potential differences between the two incidents when determining not to include welds No. 3 and No. 4 in the evaluation and documentation of the generic implications of the root and contributing causes; and therefore, did not provide a justification for excluding welds No. 3 and No.4 from this evaluation or corrective actions.

The inspectors identified that the licensee had not followed procedure EN-LI-118, in the root cause review of the CRDM-24 leak as documented in report CR-PLP-2013-05623. Section 5.5 (12)e of EN-LI-118 required that the licensee "perform an extent of cause evaluation by reviewing the individual root and contributing causes for generic implications to establish whether the causes can affect other SSCs." Additional details are provided in the procedure on how to conduct and document the evaluation. In this case, the inspectors identified that the licensee had not addressed or documented a basis in RCR CR-PLP-2013-05623 to exclude welds No. 3 and No. 4 from the generic factors discussed above that led to the 2012 leak in CRDM-24 (e.g., TGSCC at weld No. 5). The licensee entered this issue into the CAP as CR-PLP-2013-01500. Subsequently, the licensee decided to revise the inspection plan to add additional corrective actions to inspect a sample of CRDM welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Analysis: The inspectors determined that the failure to adequately evaluate and document the generic implications of the cause of the cracking identified in CRDM-24 in accordance with the root cause procedure EN-LI-118 was a performance deficiency that warranted a significance evaluation. The inspectors determined that this issue was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the Initiating Events Cornerstone attribute of equipment performance and adversely affected 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. The inspectors also answered "yes" to the More-than-Minor screening question, "if left uncorrected, would the performance deficiency have the potential to lead to a more significant safety concern"? Specifically, absent NRC identification, the licensee would not have completed further evaluations or inspections of CRDM housing welds, which could have resulted in additional CRDM housing failure and leakage by TGSCC. In accordance with Table 2 "Cornerstones Affected by Degraded Condition or Programmatic Weakness," of IMC 609, Attachment 4, "Initial Characterization of Findings," issued June 19, 2012, the inspectors determined that the Initiating Events Cornerstone was affected because the failure of a CRDM housing is a Primary System LOCA initiator contributor.

The inspectors determined this finding was of very low safety significance (Green) based on answering "no" to the Exhibit 1, "Initiating Events Screening Questions," in IMC 0609, Attachment A, "The Significance Determination Process (SDP) for Findings At-Power" issued on June 19, 2012. Specifically, the inspectors answered "no" to the screening question associated with exceeding the reactor coolant system leak rate for a small LOCA and "no" to the question associated with whether the finding could have likely affected other systems used to mitigate a LOCA resulting in a total loss of their function. The inspectors answered no to these questions because of the inherent toughness (e.g., flaw tolerance) of the Type 316 stainless steel material is such that leakage rates well below a small break LOCA would be observed through inservice cracks and actions taken to correct them prior to experiencing a large component rupture.

The inspectors determined that the primary cause of the failure to adequately consider welds No. 3 and No. 4 in the generic implications section of the RCR related to the cross-cutting component of decision making the area of human performance because licensee staff did not use conservative assumptions in decision making. Specifically, the licensee did not use conservative assumptions when excluding welds No. 3 and No. 4 as being susceptible to TGSCC; the licensee should have included them in the generic implications section of the RCR. (Item H.1(b) of IMC 310).

Enforcement: The inspectors identified one violation of NRC requirements that was associated with this finding:

- Title 10 CFR Part 50, Appendix B, Criterion V "Instructions, Procedures and Drawings, requires in part, activities affecting quality shall be prescribed by documented procedures and shall be accomplished in accordance with these procedures.

Procedure EN-LI-118, "Root Cause Evaluation Process," Revision 17, states:

- Section 5.5 (12)e perform an extent of cause evaluation by reviewing the individual root and contributing causes for generic implications to establish whether the causes can affects other SSCs, organizations or work processes. Use the Two-Step Process in accordance with Attachment 9.7.

- Attachment 9.7: Determine whether the occurrence/consequence (problem) is isolated, or whether it has broader (generic or common mode) implications. Achieve this by asking the following questions:

- Could this happen to equipment that is similar in function, design, or service condition?
- Could this happen to a group of components? (components of the same construction or materials that could be similarly affected by one condition)?

- Attachment 9.7: Document the results of the above considerations. Include the following items in the write up:

  - Generic Implications. (Is this problem/cause limited to this component/equipment, or does it apply to others as well)?
  - Existing broader (generic/common mode) considerations.

- Section 5.5(15)(10)c and f: Document proposed corrective actions and due dates to address valid generic implications. If no corrective action is recommended for a valid generic implication then document the basis for this conclusion and any risk or consequence identified, as a result of taking no action.

Contrary to the above, from February 24, 2013, through April 18, 2013, the licensee failed to accomplish activities affecting quality in accordance with procedure EN-LI-118, Revision 17, which was being implemented to correct a SCAQ. Specifically, the licensee failed to accomplish Section 5.5 (12)e by not fully evaluating and documenting the existing broader (generic/common mode) considerations, extent of condition/cause associated with TGSCC at CRDM housing welds No. 3 and No. 4, including considering the susceptibility of the welds to TGSCC and the need to perform subsequent inspections or evaluations.

Subsequently, the licensee revised the inspection plan to add additional corrective actions to inspect a sample of welds No. 3 and No. 4 for TGSCC during the upcoming refueling outage.

Because of the very low safety significance and because the licensee entered this issue into their CAP as CR-PLP-2013-01500, the violation is being treated as an NCV consistent with Section 2.3.2 of the Enforcement Policy (NCV 05000255/2013003-06; Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24).

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2010004-06: Use of TLDs May Not Be Consistent With the Methods Used By the Nation Voluntary Laboratory Accreditation Program Accreditation Process

In the third quarter of 2010 the inspectors identified that the licensee's use of thermoluminescent dosimeters (TLDs) may not be consistent with the methods used by the NVLAP accreditation process. Specifically, the licensee used a vendor to supply and process dosimeters that measure radiation exposure for the monitored workers. This vendor is NVLAP-accredited for beta, gamma, neutron, mixture of beta/gamma, and mixture of neutron/gamma radiations. However, the licensee used the TLDs when workers may be exposed to beta, gamma, and neutron radiations within the same monitoring period. The inspectors determined that this mixture of three radiation types may not be aligned with the accreditation process and opened URI 05000255/2010004-

06 to evaluate the issue. The inspectors requested technical assistance from the Office of Nuclear Reactor Regulation through Task Interface Agreement 2012-05 (ML 12268A330), the results of which, are discussed below.

Title 10 CFR 20.1501(c)(2) requires that the dosimeter processor be approved for the type of radiation or radiations included in the National Voluntary Laboratory Accreditation Program (NVLAP) Program that most closely approximates the type of radiation or radiations for which the individual wearing the dosimeter is monitored. As there is no NVLAP test category for dosimeters exposed to a mixture of beta, gamma, and neutron radiations, the NRC has determined that licensees, which monitor for beta, gamma, and neutron exposure with a single dosimeter need to use a processor that is NVLAP accredited in categories for beta-photon mixtures and neutron-photon mixtures. The licensee's dosimetry processor was NVLAP accredited for both beta-photon and neutron-photon mixtures and therefore was in compliance with 10 CFR 20.1501(c)(2).

Notwithstanding the paragraph above, licensees are required to provide adequate monitoring in accordance with 10 CFR 20.1502(a). For any type of in-field use practice that can introduce error in the monitoring results (dependent upon type of dosimeter and processing method), it becomes a question of compliance with the monitoring requirements of 10 CRR 20.1502(a) and not of NVLAP accreditation requirements of 10 CFR 20.1501(c)(2). As described in TIA 2012-05, the licensee had performed a study with the dosimeter type used at the facility (Harshaw 760). This study showed that exposing a single Harshaw 760 dosimeter to a mixture of beta, gamma, and neutron radiation met industry standard for accuracy and precision. Therefore, the licensee provided adequate monitoring and was in compliance with 10 CFR 20.1502(a).

The inspectors determined that no performance deficiency existed; therefore, this URI is closed.

.2 <u>(Closed) Unresolved Item 05000255/2012012-02: Potential Inadequate Degradation Evaluation of Control Rod Drive Mechanism Housings (This inspection is part of the additional inspections included in the Palisades Deviation letter)</u>

During a special inspection performed in August 2012, NRC inspectors identified an issue, which could not be resolved without additional information (URI). This issue was associated with the rate of growth of the crack, which created the through wall leak in CRDM-24, discovered on August 12, 2012. Identification of this crack growth rate is significant in determining appropriate intervals for future inspections to provide reasonable assurance that CRDM housing leakage will not recur.

Preliminary failure analysis data available during the inspection indicated that the observed cracking was due to TGSCC. Cracking of this type is normally due to the presence of oxygen and chlorides at the location of the crack. When examining the fracture surface at the location the through-wall leak occurred, the licensee identified six concentric rings (beach marks) propagating in a radial direction from the ID out towards the OD of the housing. Beach marks are normally associated with fatigue failures and indicate the number of stress cycles from crack initiation to crack failure. In this case, there was no evidence that fatigue contributed to the failure. Despite the lack of evidence of fatigue, it was apparent that the crack, which resulted in the CRDM-24 leak, grew in increments. It was not, however, immediately apparent whether the increments

were related to oxygen ingress (refueling outages) or temperature/pressure cycles (heatups/cooldowns).

At the time of the original inspection, five time intervals for through-wall crack growth were under consideration. Two were based on literature crack growth data and three were based on interpretations of the beach marks. These time intervals were:

- Based on literature data, one contractor estimated that a 10 percent through-wall flaw would require 4 years to reach 50 percent through-wall.
- Based on literature data another contractor estimated the crack growth rate to be 2.1×10^{-5} inches/hour or 0.18 inches/year. This is approximately three times faster than the crack growth rate proposed in the above mentioned rate.
- Based on the concept of oxygen ingress at refueling outages, six cycles of 18 months duration would require 9 years for the crack to grow through wall.
- Based on the concept of temperature/pressure cycles, the plant experienced six cold shutdowns in approximately two years preceding the crack. This equates to 2 years for the crack to grow through wall.
- Based on the concept that oxygen is required for crack growth and that oxygen is rapidly purged from the CRDM housings due to leakage past the seals, crack growth occurs only during the first few weeks of operation following a refueling outage, followed by no growth for the remaining period of operation when oxygen concentrations are low. This equates to six oxygen ingress events (irrespective of time between events) for the crack to grow through wall.

NRC inspectors including technical experts from NRC Headquarters performed a follow up inspection to determine if the assumptions made by the licensee were conservative and the planned actions bounded those conservative assumptions. The inspectors reviewed a variety of documents associated with crack growth and inspection intervals. The inspectors noted the following statements included in the RCR and vendor documents related to the determination of the appropriate crack growth rate:

- The laboratory conducting the failure analysis concluded, it could not be conclusively determined if the beach marks corresponded to refueling outages, (i.e., 18 month cycle) or shorter periods as occurred during outages over the past 24 months.
- Palisades CRDM-21 leaked at weld No. 3 in 2001. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 failure. In calculating the crack growth rate of this crack, one contractor utilized an interval between beach marks, which is much shorter than refueling outages. The intervals used are consistent with plant thermal cycles in which oxygen may or may not have been admitted into the CRDMs.
- A CRDM housing at Fort Calhoun leaked at weld No. 5 in 1990. The fracture surface of the crack leading to this leak contained beach marks identical to those in the 2012 Palisades failure. In calculating the crack growth rate of this crack, Fort Calhoun stated that the beach marks were related to refueling cycles. Fort Calhoun also performed calculations indicating that the oxygen level at the location of the flaw did not change with time (including in response to refueling outages) because the CRDM housing was not vented. Fort Calhoun's evaluation indicated that oxygen levels at the vicinity of the crack would have begun to decline through diffusion and convection had the intervals between outages been

much longer than 18 months. This is interpreted to mean that the beach marks at Fort Calhoun are in response to pressure/thermal cycles.

- In at least one instance, Palisades needed to repair the seals on a reactor coolant pump at a time other than an outage. This necessitated draining some of the water from the reactor coolant system and venting (admitting oxygen into) the CRDM housing. This represented an additional oxygen ingress event not included when determination of time to cracking is based on refueling outages.
- In its inspection plan, Palisades stated that it will inspect all CRDM housings over the next four refueling outages, i.e., the interval between inspections is one refueling outage.

Based on the above review, the inspectors noted that there were certain non-conservative statements contained in the RCR and the inspection plan. These included:

- The crack growth rate based on refueling outages was understated. If oxygen ingress is related to beach marks, given the oxygen ingress event, which occurred to repair reactor coolant pump seals, six beach marks would occur in a maximum of five refueling intervals rather than the six refueling intervals that were used to calculate the crack growth rate in the RCR.
- The crack growth rate based on heat up and cool down cycles is overstated. The value in the root cause is based on 11 months. While six shutdowns did occur at the plant in 11 months several of these events did not result in pressure/ temperature changes of the reactor coolant system. The appropriate timeframe is 24 months rather than 11.
- The inspection plan contains a non-conservative statement: "However, once the crack has been initiated it propagates over four to five operating cycles prior to going through wall." While this statement does reflect one of the proposed theories for crack growth, sufficient evidence to demonstrate reasonable assurance that this theory is correct, and thereby overcome the non-conservatism of this statement, was not provided.

Despite the existence of the non-conservatisms stated above, the inspectors concluded:

- Sufficient evidence to conclusively determine the rate of crack growth does not exist.
- Crack growth based on pressure/temperature cycles is the most conservative of the potential crack growth mechanisms. In the absence of reasonable assurance of the correctness of less conservative mechanisms, through wall crack growth in two years must be utilized for regulatory purposes;
- The licensee has not formally committed to any of the crack growth mechanisms discussed.
- The licensee's inspection program includes inspection of all of the CRDM housings over the next four refueling outages. Approximately 25 percent of the housings will be inspected during each outage. The inspection of 25 percent of the CRDM housings each interval is sufficient to indicate that, in the event no indications are found during a given inspection, the probability that flaws exist in other housings is extremely low. As such, it may be considered that the inspection of approximately 25 percent of the CRDM housings every refueling outage bounds all the crack growth rate mechanisms considered.

Overall, some weaknesses did exist in the site's assessment, but none of these issues arose above the level of a minor performance deficiency for the evaluations completed. With the corrective actions in place to monitor the CRDMs, the inspectors considered this approach to inspection to be both acceptable and sufficient justification to close this URI.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 22, 2013, the inspectors presented the inspection results to Mr. A. Vitale and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The inspection results regarding the biennial written and annual operating test and biennial review inspection on January 10, 2013, to Mr. A. Vitale, Site Vice President, and other members of the licensee staff. The licensee acknowledged the issues presented.
- The inspection results for the areas of occupational dose assessment; and radioactive solid waste processing and radioactive material handling, storage, and transportation with Mr. A. Vitale, Site Vice President, and other members of the licensee staff on February 15, 2013.
- The inspection results for the triennial heat sink inspection with Mr. A. Vitale, Site Vice President, and other members of the licensee staff on April 18, 2013.
- The results of the problem identification and resolution in-depth sample on CRDM leakage with Mr. A. Vitale, Site Vice President, on April 18, 2013.

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 or destroyed.

4OA7 Licensee-Identified Violations

The following violation of very low significance (Green) or Severity Level IV was identified by the licensee and is a violation of NRC requirements which meets the criteria of the NRC Enforcement Policy for being dispositioned as an NCV.

- One licensee identified finding of very low safety significance and an associated NCV was reviewed by the inspectors. Title 10 of the CFR, 71.5 (a), requires, in part, that each licensee who transports licensed material outside the site of usage, as specified in the NRC license, or where transport is on public highways, or who delivers licensed material to a carrier for transport, shall comply with the applicable requirements of the Department of Transportation Regulations in 49 CFR 107, 171 through 180, and 390 through 397, appropriate to the mode of transport. Title 49 CFR, 172.202, requires, in part, that descriptions of hazardous material on shipping papers include the identification number prescribed for the material and the proper shipping name prescribed for the material.

Contrary to the above, on May 20, 2011, the licensee failed to use the proper shipping name and proper UN (United Nations) number when shipping radioactive resin in a Type B Cask. This issue was documented in the licensee's CAP in CR-PLP-2011- 02887. Immediate corrective actions included notifying appropriate personnel and correcting the shipping records. The finding was assessed in accordance with IMC 0609, Attachment D, Public Radiation Safety Significance Determination Process, and determined to be of very low safety significance (Green). Specifically, the inspectors determined that the finding did not involve the Radioactive Effluent Release Program or the Radiological Environmental Monitoring Program. The finding did involve the transportation of radioactive material. However, no external radiation levels or surface contamination levels were exceeded, the finding did not involve the certificate of compliance, there was no failure to make notifications or provide emergency information, there was no breach of the package during transit, and there was no low-level burial ground non-conformance.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

A. Vitale, Site Vice President
T. Williams, General Plant Manager
C. Amone, NSA Director
B. Baker, Assistant Operations Manager
M. Cimock, Programs Engineer
B. Davis, Engineering Director
B. Dotson, Licensing
T. Foudy, Engineering Supervisor
T. Groth, Quality Assurance
O. Gustafson, Licensing Manager
J. Hager, Program Owner
T. Horan, Training Superintendent
D. Malone, Emergency Preparedness Manager
J. Milliken, Programs Engineering Supervisor
S. Mims, Radiation Protection Supervisor
T. Mulford, Assistant Operations Manager
B. Nixon, Training Manager
C. Smith, Security Superintendent
J. Smith, Radiation Protection Shipper
B. VanWagner, DFS Manager
J. Walker, Quality Assurance Manager
D. Watkins, Radiation Protection Shipper
B. Williams, Engineer

Nuclear Regulatory Commission

G. Hansen, Security Inspector
M. Jones, Reactor Inspector
A.M. Stone, Chief Engineering Branch 2
T. Taylor, Senior Resident Inspector
R. K. Walton, Senior Operator Licensing Inspector
C. Zoia, Operator Licensing Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000255/2013002-01	NCV	Failure to Establish an Acceptable Component Cooling Water Heat Exchanger Final Test Frequency (1R07)
05000255/2013002-02	NCV	Inadequate Work Instructions for Component Cooling Water Heat Exchanger (1R18)
05000255/2013002-03	NCV	Damage to ‘A’ AFW Pump Packing During Surveillance Run (1R22)

05000255/2013002-04	NCV	Failure to Perform DAC-Hour Tracking (2RS4)
05000255/2013002-05	NCV	Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage (4OA2.4(b.1))
05000255/2013002-06	NCV	Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24 (4OA2.4(b.2))

Closed

05000255/2013002-01	NCV	Failure to Establish an Acceptable Component Cooling Water Heat Exchanger Final Test Frequency (1R07)
05000255/2013002-02	NCV	Inadequate Work Instructions for Component Cooling Water Heat Exchanger (1R18)
05000255/2013002-03	NCV	Damage to 'A' AFW Pump Packing During Surveillance Run (1R22)
05000255/2013002-04	NCV	Failure to Perform DAC-Hour Tracking (2RS4)
05000255/2012005-03	URI	Safety Injection Refueling Water Tank Evaluation of Corrosion (4OA2.3)
05000255/2013002-05	NCV	Failure to Take Corrective Action to Prevent Recurrence of CRDM Pressure Boundary Leakage (4OA2.4(b.1))
05000255/2013002-06	NCV	Failure to Adequately Address the Generic Implications of the Cracking Identified in CRDM-24 (4OA2.4(b.2))
05000255/2012012-01	URI	TS for PCS Pressure Boundary Leakage (4OA2.4)
05000255/2012012-03	URI	Potential Failure to Prevent Recurrence of a Significant Condition Adverse to Quality (4OA2.4)
05000255/2010004-06	URI	Use of TLDs May Not Be Consistent With the Methods Used By the NVLAP Accreditation Process (4OA5.1)
05000255/2012012-02	URI	Potential Inadequate Degradation Evaluation of CRDM Housings (4OA5.2)

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.

1R04 Equipment Alignment

- CR-PLP-2013-00873, NRC-identified: CV-0847 A/S SV-0847 Vent, was not properly locked in its checklist-required locked-open position, February 26, 2013
- DBD 1.01, Component Cooling Water System, Revision 8
- Design Basis Document-1.02, Service Water System, Revision 8
- Design Basis Document-1.06, Control Room Heating and Ventilation System, Revision 8
- M-208, P&ID Critical Service Water System, Sheet 1A, Revision 62
- M-208, P&ID Non-Critical Service Water System, Sheet 1, Revision 104
- M-218, P&ID Heating, Ventilation, and Air Conditioning Control Room, Sheet 6, Revision 16
- SOP-12, Auxiliary Feedwater System Checklist, Revision 68
- SOP-15, Service Water System, Revision 54
- SOP-16, Component Cooling Water System, Revision 38
- SOP-24, Ventilation and Air Conditioning System, Revision 58
- SWS-E-5, Service Water Pump Motor Removal, Maintenance, and Replacement, Revision 19
- WO #52438269, P-7A Replacement of Packing Sleeve, February 25, 2013

1R05 Fire Protection

- EA-FPP-03-001, Fire Protection Calculation, Revision 3
- Fire Hazards Analysis, Revision 7
- Fire Protection Compensatory Actions, January 15, 2013
- Palisades Auxiliary Building – Cable Spreading Room and 1-D Switchgear Room / Elev. 607' Pre-Fire Plans (Fire Areas 2 & 3)
- Palisades Auxiliary Building – Diesel Generator 1-1 and Fuel Oil Day Tank Rooms / Elev. 590' Pre –Fire Plan (Fire Areas 5 & 7)
- Palisades Auxiliary Building – West Engineering Safeguards Room / Elev. 570' Pre-Fire Plan (Fire Area 28)
- Palisades Turbine Building – General Areas / Elev. 625'/612'/607' Pre-Fire Plan (Fire Area 23)

1R06 Flood Protection Measures

- CR-PLP-2011-6818, NRC resident questioned equipment that could become wetted, December 14, 2011
- DBD 7.08, Plant Protection from Flooding, Revision 6
- FSAR Chapter 5, Design of Structures, Systems, and Components, Revision 29

1R07 Heat Sink Performance

- ARP-7, Auxiliary Systems Scheme EK-11 (C-13), Revision 83
- Audit and Review Report for Plant Aging Management Reviews and Programs, October 20, 2005
- Component Cooling Heat Exchanger Technical Justification 2010
- Component Cooling Heat Exchanger Technical Justification 2012

- Component Cooling System Health Report, Q1-2012
- COP-16A, COMPONENT COOLING WATER SYSTEM CHEMISTRY, Revision 16
- Critical Service Water System Health Report, Q1-2012
- CR-PLP-2011-07085, Service flow to CCW Heat Exchanger may be limited, December 28, 2011
- CR-PLP-2012-04423, Sand Found in Cooling Water Basins, October 5, 2010
- CR-PLP-2012-05111, Electrical Conduit Partially Discolored from water from P-7A, July 17, 2012
- CR-PLP2012-05111, P-7A Packing Leakage in contact with Electrical Housing, July 17, 2012
- CR-PLP-2012-05112, Oil in P-7C lower reservoir slightly discolored, July 17, 2012
- CR-PLP-2012-05114, Corrosion deposit on CCW hx flange, July 17, 2012
- CR-PLP2012-05114, Corrosion Deposit on Service Water Side of E-54A, July 17, 2012
- CR-PLP2012-05115, Ladders in CCW Room Scaffolding Lay Down Area in contact with Nitrogen Supply Lines, July 17, 2012
- CR-PLP-2012-05115, Plant Ladder sitting on Nitrogen Lines, July 17, 2012
- CR-PLP2012-0512, Slight discoloration in oil in P-7C Lower Oil Reservoir, July 17, 2012
- CR-PLP-2012-05132, CCW Heat Exchanger frequency, July 18, 2012
- CR-PLP-2012-05132, NRC Question Technical Justification for extending inspection frequency, July 18, 2012
- E-54A inspection report ENO1-PN1-01 dated September 15, 2007
- E-54A inspection report ENO4-PN1-01 dated April 5, 2009
- EA-C-PAL-99-1209B-01, Generation of Flow Rate Acceptance Criteria, Revision 3
- EA-GOTHIC-04-05, Development of LOCA Containment Response Base Deck, Revision 1
- EN-DC-136, Temporary Modifications, Revision 7
- EN-DC-184, NRC Generic Letter 89-13 Service Water Program, Revision 2
- EN-DC-600, Long Term Asset Management Plan, February 14, 2012
- IN 94-003 Deficiencies Identified During Service water Systems Operational Performance Inspection
- Installation of NO Boater Buoys and Inspection of Intake Structure, June 7, 2012
- LIT-1336A, Instrument Calibration Sheet, November 1, 2011
- LIT-1336B, Instrument Calibration Sheet, October 15, 2009
- M-208, P&ID Service Water System 1A, Revision 62
- M-208, P&ID Service Water System 1B, Revision 36
- M-208, Piping and Instrument Diagram Service Water System, Revision 1A
- M-209, P&ID Component Cooling System sht. 1, Revision 68
- M-209, P&ID Component Cooling System sht. 2, Revision 33
- M-209, P&ID Component Cooling System sht. 3, Revision 52
- M-213, P&ID Service Water, Screen Structure and Chlorinator, Revision 93
- ONP 6.1, Loss of Service Water, Revision 16
- ONP-6.1, Loss of Service Water, Revision16
- PAD11-0182, EC-29096 Assess the Viability of Selected Replacement Packing Styles in Service Water Pumps P-7B and P-7C, Revision 0
- Procedure CCS-M-2, Component Cooling Water Heat Exchanger Maintenance, Revision 22
- RO-216, SERVICE WATER FLOW VERIFICATION, Revision 17
- SEP-SW-PLP-001, Raw Water Corrosion Program, Revision 0
- SOP-16, COMPONENT COOLING WATER SYSTEM, Revision 17
- T-390, SINGLE TUBE TESTING OF THE CCW HEAT EXCHANGERS, Revision 4
- VEN-M14, E-54A tube plugging map, Sheet 26, Revision 6
- VTD-0549-0004, Industrial Process Engineers Instruction Manual for Component Cooling Heat Exchangers E-54-A & E-54B
- WO 265456, E-54A, Clean CCW HX tubes

1R11 Licensed Operator Requalification Program

- Administrative Procedure 4.00, “Operations Organization, Responsibilities and Conduct,” Revision 49
- EAL Basis Document, Revision 5
- Emergency Operating Procedure Supplement 1, Pressure Temperature Limit Curves, Revision 5
- EOP-1, Standard Post Trip Actions, Revision 13
- GOP-2, Mode 5 to Mode 3 ≥525°F, Revision 35
- GOP-3, Mode 3 >/= 525 deg to Mode 2, Revision 31
- Operations Proficiency Watch Record 2012, 3rd and 4th Quarters
- Operations Proficiency Watch Record 2013, 1st Quarter
- PO-2, Technical Specification Surveillance Procedure: PCS Heatup/Cooldown Operations, Revision 6
- Simulator Exam Scenario 115

1R12 Maintenance Effectiveness

- “C-2B Instrument Air Compressor Action Plan,” Revision 2
- “System Health Report: Instrument Air System,” February 13, 2013
- Calculation EA-ELEC-VOLT-050, MCC Motor Control Circuit Voltage Analysis
- CR-PLP-2010-00132, Undervoltage pickup value for Citation motor starters incorrect, January 12, 2010
- CR-PLP-2011-03406, 52-1406 screenwash pump breaker open with springs discharged, July 11, 2011
- CR-PLP-2011-03766, Ultimate heat sink exceeded maintenance rule performance criteria, August 1, 2011
- CR-PLP-2011-03777, Maintenance Rule (a)(1) Action Plan for Instrument Air System, Revision 3
- CR-PLP-2012-06695, C-2A Instrument Air Compressor Tripped on Second Stage Inlet High Temperature, October 14, 2012
- CR-PLP-2012-07386, C-2C Instrument Air Compressor Tripped on High First Stage Discharge Temperature, November 27, 2012
- CR-PLP-2012-07420, During Troubleshooting on Instrument Air Compressor C-2C Found Wires for 2ATT (First Stage Discharge Temperature) and 4ATT (Second Stage Discharge Temperature) Swapped, November 27, 2012
- CR-PLP-2012-07443, Found Same Wiring Discrepancies in All Three Instrument Air Compressors, November 28, 2012
- CR-PLP-2012-07530, Instrument Air System Experience Additional Functional Failure While in (a)(1) Status, December 4, 2012
- CR-PLP-2012-07644, C-2A Instrument Air Compressor Availability Has Fallen Below the Maintenance Rule Monitoring Goal, December 11, 2012
- CR-PLP-2012-07733, C-2B Instrument Air Compressor Tripped on Pressure Transducer Failure, December 14, 2012
- CR-PLP-2013-00044, 5 failures in past 15 months of K-line breakers, January 4, 2013
- CR-PLP-2013-00116, Instrument Air System Experienced Additional Functional Failure While in (a)(1) Status, January 10, 2013
- CR-PLP-2013-00320, Part Removed from Instrument Air Compressor C-2B was not Received by Warehouse, January 24, 2013
- CR-PLP-2013-00529, While researching NRC question, identified breaker overhaul with incorrect due date, February 6, 2013

- EN-DC-205, Maintenance Rule Monitoring, Revision 4
- EN-DC-206, Maintenance Rule (a)(1) Process, Revision 2
- EN-DC-324, Preventative Maintenance Program, Revision 8
- EN-MA-125, Troubleshooting Control of Maintenance Activities, Revision 12
- Higher-Tier Apparent Cause Evaluation: C-2C Instrument Air Compressor Tripped on First Stage Discharge High Temperature, Revision 2, January 14, 2013
- LO-PLPLO-2009-00128, Breaker Program Focused Self-Assessment, May 18, 2010
- M-212, P&ID Service and Instrument Air System, Sheet 1A, Revision 2
- SPS-E-11, Attachment 3, Contactor Pickup Voltage, Revision 23
- Various system health reports: 480V system
- VEN-M41, Schematic and Wiring Diagram for Instrument Air Compressors C-2A, C-2B, & C-2C, Sheet 1, Revision 4
- WR #291896, C-2C Instrument Air Compressor Tripped on High First Stage Discharge Temperature, November 27, 2012

1R13 Maintenance Risk Assessments and Emergent Work Control

- Administrative Procedure No 4.28, Control of Palisades Switchyard Activities, Revision 5
- Administrative Procedure-4.02, Risk Management and Risk Monitoring, Revision 62
- ARP-1, Turbine High Vibration, Revision 69
- CR-PLP-2012-07864, Turbine supervisory panel power supply appears to be failing, December 29, 2012
- CR-PLP-2013-00337, Scheduled Task to Transfer a Component into West Engineered Safeguards via the 590' Aux Building Floor Plug was Challenged, January 24, 2013
- CR-PLP-2013-00337, Work to remove west safeguards floor plugs challenged, January 25, 2013
- CR-PLP-2013-00494, During Infrared Survey Conducted as Part of ODMI for MOD 26H5, a Hot Spot was Identified on the Z Phase of Greater than 250 Degrees F, February 4, 2013
- CR-PLP-2013-00562, During Infrared Survey Conducted as Part of ODMI for MOD-26H5, a Hot Spot was Identified on the Z Phase of Greater than 300 Degrees F, February 9, 2013
- CR-PLP-2013-00576, During Infrared Survey Conducted as Part of ODMI for MOD-26H5, a Hot Spot was Identified on the Z Phase of Greater than 300 Degrees F, February 11, 2013
- CR-PLP-2013-00586, During Infrared Survey Conducted as Part of ODMI for MOD-26H5, a Hot Spot was Identified on the Z Phase of Greater than 300 Degrees F, February 11, 2013
- CR-PLP-2013-00595, During Infrared Survey Conducted as Part of ODMI for MOD-26H5, a Hot Spot was Identified on the Z Phase of Greater than 300 Degrees F, February 12, 2013
- CR-PLP-2013-00654, During Infrared Survey Conducted as Part of ODMI for MOD-26H5, a Hot Spot was Identified on the Z Phase of Greater than 300 Degrees F, February 15, 2013
- CR-PLP-2013-01151, PI-1489, Starting Air Pressure Indicator, Was Found Indicating Pressure After Being Tagged Out, March 18, 2013
- CR-PLP-2013-01153, Prior to Racking Breaker for 1-2 EDG Out Of Cubicle, Found Wires Showing Signs of Physical Wear, March 18, 2013
- CR-PLP-2013-01166, During Preparation Activities for 1-2 EDG Maintenance Window, A Change was made to the Tagout for the Air Start System Without Verifying the Boundaries First, March 18, 2013
- CR-PLP-2013-01207, While Performing Second/Third Cycle Maintenance PM on 1-2 EDG Found Compressor Impeller to Casing Side Clearances Outside of Vendor Listing, March 20, 2013
- CR-PLP-2013-01245, During EDG Run, a Leak on the East End Bell of the Jacket Water Heat Exchanger was Identified, March 21, 2013
- Design Basis Document-1.07, Auxiliary Building HVAC Systems, Revision 5

- EN-OP-119, Protected Equipment Postings, Revision 5
- EN-WM-104, On-Line Risk Assessment, Revision 6
- EPS-M-14A, Permanent Maintenance Procedure: Diesel Generator Every Cycle Maintenance, Revision 2
- EPS-M-14B, Permanent Maintenance Procedure: Diesel Generator Second/Third Cycle Maintenance, Revision 0
- FSAR Section 14.22, Maximum Hypothetical Accident, Revision 28
- M-209, P&ID Component Cooling System, Sheet 1, Revision 68
- M-209, P&ID Component Cooling System, Sheet 2, Revision 33
- M-209, P&ID Component Cooling System, Sheet 3, Revision 53
- M-209, System Diagram: Component Cooling System, Sheet A, Revision 8
- M-218, P&ID Heating, Ventilation, and Air Conditioning Radwaste Area, Sheet 4, Revision 26
- NRC Letter, Correction Letter for Alternate Radiological Source Term Amendment (TAC MD3087)
- ONP-6.2, Loss of Component Cooling, Revision 11
- Operational Decision-Making Issue: MOD-26H5 Main Generator Disconnect Hot Spot, Revision 0?
- Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000
- RT-71L, TS Pressure Test of ESS Pump Suction Piping, Revision 19
- SOP-16, Component Cooling Water System, Revision 38
- SOP-22, Emergency Diesel Generators, Revision 57
- SOP-8, Main Turbine and Generating Systems, Revision 92
- Tagout 1C23-1 CCS-022-E-54A
- V 13820-49, Vendor Drawing for Vertical Break Switch Stationary Contact, Revision K
- V 36220-22135, Vendor Drawing for Vertical Break Switch Phase, Revision B
- VEN-M3A, Cabinet Terminal Blocks TB36-TB39, Revision 9
- Westinghouse Operation and Maintenance Memo 108, Main Stop and Reheat Valves, February 9, 1990
- WO #337289, Replace Turbine Supervisory Instrument Power Supply, P/S-0516B, January 22, 2013
- WO #338570, MOD-26H5 Repair Hot Spot at Switch Z Phase Contact Area, February 12, 2013
- WO #52370209, 24 Month PM of 1-2 EDG Engine and Fuel System, March 20, 2013
- WO 313521, Adjust Reheat and Stop Valves
- WO 337289, Replace turbine supervisory panel power supply

<u>1R15 Operability Determinations and Functionality Assessments</u>

- As-Found Frequency Test Data for 1-1 and 1-2 Emergency Diesel Generators, November 2002 through December 2012
- CR-PLP-2009-02006, Condition Report to Track the Potential Need to Revise Technical Specifications Associated with the Maximum Allowed Emergency Diesel Generator Steady State Frequency, April 12, 2009
- CR-PLP-2012-07617, PCP impellers manufactured to wrong size, December 7, 2012
- CR-PLP-2013-00295, Plywood Installed to Keep EDG Vent Fan Operable, January 22, 2013
- CR-PLP-2013-00302, Winding Insulation Damage Identified on V-24D EDG Vent Fan During Maintenance, January 22, 2013
- CR-PLP-2013-00503, CCW Surge Tank (T-3) is Showing a Lowering Trend, February 5, 2013
- CR-PLP-2013-00568, CCW Surge Tank (T-3) is Showing a Lowering Trend around 0500 on February 9, 2013, February 10, 2013

- CR-PLP-2013-00588, Maintenance Identified that V-24D EDG Vent Fan Blade Pitch is Different from Old Part Compared to New Part, February 11, 2013
- CR-PLP-2013-00652, Entered LCO 3.7.7 Due to Indications of a CCW Leak on 'A' CCW Heat Exchanger, February 14, 2013
- Design Basis Document-1.07, Auxiliary Building HVAC Systems, Revision 5
- EA-MOD-2005-004-03, ESS Recirculation Mode Flows and NPSH, Revision 4
- EC #31817, Revise Design Basis Pump Curves Using Most Recent Pump Test Data and Degradation Methodology for ECCS Pumps, Revision 0
- EC #42310, Basis TMod Evaluation for Procedurally Controlling the Use of a Temporary Mechanical Block on Out-Of-Service EDG Vent Fan to Maintain EDG Vent System Functional During EDG Vent Fan Maintenance
- EN-DC-136, Temporary Modifications, Revision 8
- EN-DC-313, Procurement Engineering Process, Revision 7
- EN-MA-125, Troubleshooting Control of Maintenance Activities, Revision 12
- EN-MP-100, Critical Procurements, Revision 9
- Failure Mode Analysis: CCW Leak, February 11, 2013
- M-209, P&ID Component Cooling System, Sheet 1, Revision 68
- M-209, P&ID Component Cooling System, Sheet 2, Revision 33
- M-209, P&ID Component Cooling System, Sheet 3, Revision 53
- M-209, System Diagram: Component Cooling System, Sheet A, Revision 8
- ONP-6.2, Loss of Component Cooling, Revision 11
- Operability Evaluation for Plywood Installed on Intake of V-24D EDG Vent Fan to Support Functionality of V-24C, January 24, 2013
- SCR-2009-0043, Software Change Request: Pipe-Flo Professional 2007a 10.01, February 12, 2009
- SOP-24, Ventilation and Air Condition System, Revision 58
- UFSAR Chapter 9, Auxiliary Systems, Section 9.3, "Component Cooling System," Revision 30
- WO #342127, Component Cooling Water Surge Tank (T-3) Indicates a System Leak of 10-20 Gallons Per Day, February 11, 2013

1R18 Temporary and Permanent Modifications

- CCS-M-2, CCW Heat Exchanger Maintenance, Revision 22
- CR-PLP-2013-00773, Enhancements for WO 265456, February 21, 2013
- CR-PLP-2013-00969, Component cooling heat exchanger tube plugging documentation shortfalls, February 20, 2013
- CR-PLP-2013-00976, CCS-M-2 section N/A'd prior to work execution, March 5, 2013
- EC 42743, Provide Design and Installation Instructions for Mechanically Driven Tubesheet Plug for CCW Heat Exchangers E-54A and E-54B
- EN-DC-115, Engineering Change Process, Revision 14
- EN-FAP-WM-011, Work Planning Standard, Revision 1
- EN-WM-105, Planning, Revision 10
- SC-86-353, Remove defective heat exchanger tubes from service by Yuba brass plugs
- WO 265456, Repair E-54A heat exchanger

1R19 Post Maintenance Testing

- ASME Section IWA 4130, Alternative Requirements
- ASME Section IWA 4540, Pressure Testing of Class 1, 2, and 3 Components
- ASME Section IWA 4700, Heat Exchanger Tubing
- CCS-M-2, CCW Heat Exchanger Maintenance, Revision 22

- CR-PLP-2013-00229, While performing WO to Replace Casing Drain Plug for P-66B Discovered that the Existing Drain Plug was Only a Little More than Hand Tight, January 18, 2013
- CR-PLP-2013-00976, CCS-M-2 section N/A'd prior to work execution, March 5, 2013
- CR-PLP-2013-01074, During Replacement of RPS Bistable for PA-0752C an Internal Switch in the Original Bistable was Found in a Different Position than Specified in the Work Order Instructions, March 12, 2013
- CR-PLP-2013-01127, RPS Bistable Loss of Signal Switch (S3) is in Non-Conforming Position for PA-0752C, March 15, 2013
- CR-PLP-2013-01131, Error Identified on RPS Bistable Trip Unit Vendor Manual During Verification of S3 Positions, March 15, 2013
- CR-PLP-2013-01195, Small Fuel Oil leak from Fuel Oil Booster Pump was Identified during 1-2 EDG PM, March 19, 2013
- CR-PLP-2013-01199, Problem Found With Gasket at the Exhaust Outlet on Top of the 1-2 EDG Turbocharger, March 19, 2013
- CR-PLP-2013-01207, While Performing Second/Third Cycle Maintenance PM on 1-2 EDG Found Compressor Impeller to Casing Side Clearances Outside of Vendor Listing, March 20, 2013
- CR-PLP-2013-01214, Minor Cracks identified in Turbocharger Supports, March 20, 2013
- CR-PLP-2013-01239, 1-2 EDG ICS Relay Coils were Found Blocked While Performing Undervoltage Start Channel Calibrations, March 20, 2013
- CR-PLP-2013-01245, During EDG Run, a Leak on the East End Bell of the Jacket Water Heat Exchanger was Identified, March 21, 2013
- CR-PLP-2013-01247, PMT for Verifying No Leakage on 1-2 EDG Overspeed Trip Device Failed, March 21, 2013
- CR-PLP-2013-01248, Minor Leak Found on Exhaust Pipe Connection at Back of 1R Cylinder Head on 1-2 EDG During Maintenance Outage, March 20, 2013
- CR-PLP-2013-01250, Emergency Diesel Generator 1-2 Failed to Start During MO-7A-2 When Using Air Start Motor-2A, March 21, 2013
- CR-PLP-2013-01273, Troubleshooting Revealed That Contact 21-22 on the Shutdown Relay of the 'A' Start Circuit Was Open When It Should Have Been Closed, March 21, 2013
- CR-PLP-2013-01275, While Replacing Failed Shutdown Relay on 1-2 EDG Dropped and Lost One of the Contact Shields, March 21, 2013
- CR-PLP-2013-01277, Found One of the Lost Contact Shields for the Shutdown Relay Lodging in the Relay and the Other Fell to the Floor, March 22, 2013
- CR-PLP-2013-01278, Found FME in Failed Shutdown Relay that Prevented the Contact from Closing, March 22, 2013
- CR-PLP-2013-01376, Service Water Flow Rate Through 1-2 EDG was Higher Than Expected During Performance of MO-7A-2, March 28, 2013
- EC #43347, RPS Bistable Loss of Signal Switch (S3) Position Guidance, Revision 0
- EN-IS-123, Electrical Safety, Revision 9
- EPS-M-14A, Permanent Maintenance Procedure: Diesel Generator Every Cycle Maintenance, Revision 2
- EPS-M-14B, Permanent Maintenance Procedure: Diesel Generator Second/Third Cycle Maintenance, Revision 0
- MO-7A-2, Technical Specification Surveillance: Emergency Diesel Generator 1-2, Revision 81
- QI-9, Technical Specification Surveillance Procedure: Reactor Protective Trip Units, Revision 15
- QO-14, Technical Specification Surveillance Procedure: Inservice Test Procedure – Service Water Pumps, Revision 35

- QO-19, Technical Specification Surveillance Procedure: Inservice Test Procedure – HPSI Pumps and ESS Check Valve Operability Test, Revision 32
- RE-136, Technical Specification Surveillance Procedure: Performance Test - Battery Charger No 4 (ED-18), Revision 8
- SOP-22, Emergency Diesel Generators, Revision 57
- SOP-30, Station Power, Revision 67
- VEN-M1-Q, Vendor Drawing 11247-47004, Bistable Trip Module Schematic, Sheet 4011, Revision 0A
- WI-EPS-E-01, Battery Charger Maintenance, Revision 10
- WI-SWS-M-03, Service Water Pump P-7A Removal, Inspection, and Reinstallation, Revision 6
- WO # 52330547, P-66B ('B' HPSI Pump) Coupling Inspection PM, January 17, 2013
- WO #00254075, P-66B Casing Drain Plugs Clean Boric Acid and Replace, January 18, 2013
- WO #00293289, CK-ES 3339 (HPSI Pump P-66B Mini-Flow Check Valve) Clean and Torque Pressure Seal, January 17, 2013
- WO #262581, Minor Gasket Oil Leak on 1-2 EDG Overspeed Trip Device
- WO #334284, Replace PA-0752C in 'C' Channel RPS Due to Failed to Illuminate Red Tell Tale Trip Light, March 12, 2013
- WO #344737, Validate Position of S3 Switch on All RPS Bistables, March 15, 2013
- WO #345493, 1-2 Diesel Generator Failed to Start During MO-7A-2, March 22, 2013
- WO #52370209, 24 Month PM of 1-2 EDG Engine and Fuel System, March 20, 2013
- WO #52374563, ED-18 Battery Charger #4 PM, February 4, 2013
- WO #52438269, P-7A Replacement of Packing Sleeve, February 25, 2013
- WO 265456, Repair E-54A heat exchanger

1R20 Outage Activities

- Component Cooling Water Heat Exchanger E-54A Tube Plugging Sheet, February 19, 2013
- GOP-14, Shutdown Cooling Operations, Revision 44
- GOP-3, Mode 3 ≥ 525°F to Mode 2, Revision 31
- GOP-8, Power Reduction and Plant Shutdown to Mode 2 or 3 ≥ 525°F, Revision 30

1R22 Surveillance Testing

- Alarm Response Procedure-21, Reactor Protective System Scheme EK-06, Revision 52
- CR-PLP-2008-03500, During Performance of QO-21, AFW Pump P-8B IST, Packing Leakoff from the Inboard Packing Gland was Less than Desired, August 13, 2008
- CR-PLP-2012-05593, Indication issue with V-2B during QO-1, August 9, 2012
- CR-PLP-2012-07693, During power ascension delta T power not calibrated and greater than 2% lower than heat balance power, December 12, 2012
- CR-PLP-2013-00070, Found Conduit Fitting disconnected from Main Header for Diesel Generator K-6A Cylinder Head Temperature, January 7, 2013
- CR-PLP-2013-00235, Green indicating light for V-2B went out during QO-1, January 18, 2013
- CR-PLP-2013-01128, P-8A Motor Driven Auxiliary Feedwater Pump was Manually Tripped Due to Smoke coming from the Inboard Packing Gland During Performance of QO-21, March 15, 2013
- DRN 11-757, Charging pump breakers
- DWO-1, Operators Daily, Weekly Items, Revision 100
- E-61, Electrical Schematic: Source and Wide Range #1/3 Nuclear Instrumentation, Sheet 2A, Revision 6
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- EA-ER-PAL-89-038C-01, Evaluation of non-conservative nuclear power level input to the RPS, January 15, 1990
- EC 21535, IST changes for CCW pumps
- Enertech letter to Palisades regarding CCW pump flow limits, March 4, 2010
- FSAR Chapter 9, Auxiliary Systems, Revision 25
- QO-21, Technical Specification Surveillance Procedure: Inservice Test Procedure – Auxiliary Feedwater Pumps, Revision 40
- RI-99, Technical Specification Surveillance Procedure: Left Channel Nuclear Instrumentation Calibrations, Revision 12
- RO-128-1, Diesel Generator 1-1 24 Hour Load Run, Revision 18
- SOP-22, Emergency Diesel Generators, Revision 56
- SOP-35, Neutron Monitoring System, Revision 16
- WO #344979, P-8A Drain/Replace/Inspect Inboard Bearing Packing, March 15, 2013
- WO #52336888, RO-128-1 Diesel Generator 1-1 24 Hour Load Run, January 7, 2013
- WO 52442713, QO-15A, ISI test for CCW pump
- WO 52450577, QO-1, Safety Injection System Test

1EP6 Drill Evaluation

- 1st Quarter Emergency Planning Integrated Drill Report for Drill Held February 27, 2013
- 1st Quarter Emergency Planning Integrated Drill, February 27, 2013
- CR-PLP-2013-0892, Players had Difficulty Logging onto the Drill RWP for Emergencies
- CR-PLP-2013-0940, Two TSC Players Did Not Have Their Assigned Dosimetry
- CR-PLP-2013-0941,TSC and EOF Leadership Declined to Approve Dose Extensions in Support of an Attempt to Isolate the Release in Progress
- EI-3, Attachment 1, Palisades Event Notification Form, Drill Message 1, February 27, 2013
- EI-3, Attachment 2, Palisades Technical Data Sheet, Drill Message 2, February 27, 2013

2RS4 Occupational Dose Assessment

- CR-PLP-2012-02683, DAC Hour Tracking and Associated Dose Assessments, October 19, 2012

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

- 2011 and 2012 Indexes of Radioactive Shipments, dated February 11, 2013
- CR-PLP 2011-05528, Shipping Records Contact Number Not Highlighted, October 21, 2011
- CR-PLP 2012-07397, Shipping Records Missing Certification Signature, November 27, 2012
- CR-PLP 2012-07591, Shipping Records Missing Reviewers Signature, December 6, 2012
- CR-PLP 2012-07593, Different Procedure Revision Documented in Shipping Records, December 6, 2012
- CR-PLP 2013-00109, Unsatisfactory Self-Assessment Results for RAM Shipping, January 09, 2013
- CR-PLP-2011-06293, Deficiencies in Regard to Completion, Review, and Processing of Quality Assurance Records. November 21, 2011
- CR-PLP-2012-04638, QA Identified SIRWT Tank Needed to be Evaluated as an Effluent Pathway, June 22, 2012
- EN-RP-121, Radioactive Material Control, Revision 07
- EN-RP-121-01, Receipt of Radioactive Material, Revision 01
- EN-RW-102, Radioactive Shipping Procedure, Revisions 10
- EN-RW-103, Radioactive Waste Tracking Procedure, Revision 03

- EN-RW-104, Attachment 9.1, 10CFR61 Part 61 Waste Stream Sample Screening and Evaluation, DAW, April 27, 2011
- EN-RW-104, Attachment 9.1, 10CFR61 Part 61 Waste Stream Sample Screening and Evaluation, T-100 Resin, May 23, 2011
- EN-RW-104, Attachment 9.1, 10CFR61 Part 61 Waste Stream Sample Screening and Evaluation, T-994 and T995 ALPS Charcoal, April 23, 2012
- EN-RW-104, Attachment 9.1, 10CFR61 Part 61 Waste Stream Sample Screening and Evaluation, T-996 ALPS Cation Resin, April 23, 2012
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- EN-RW-104, Attachment 9.1, 10CFR61 Part 61 Waste Stream Sample Screening and Evaluation, 2012 Dry Active Waste (DAW), November 01, 2012
- EN-RW-104, Scaling Factors, Revision 08
- EN-RW-105, Process Control Program, Revision 02
- Focused Self-Assessment, LO-PLPLO-2011-00087, RAM Shipping, November 22, 2012
- HP 6.18, Low-Level Radioactive Material/Waste Packaging, Revision 30
- Lesson Plan WMG RC-102, Use of WMG Programs and Regulatory Interfaces Intermediate
- Lesson PlanPL-166140, DOT Hazmat, Revision 00
- Level Course, Revision 07
- PLP-2011-RM-024, Radioactive Shipping Package Documentation, April 21, 2011
- PLP-2011-RW-010, Radioactive Shipping Package Documentation, May 20, 2011
- PLP-2011-RW-011, Radioactive Shipping Package Documentation, June 9, 2011
- PLP-2012-RM-058, Radioactive Shipping Package Documentation, May 17, 2012
- PLP-2012-RW-018, Radioactive Shipping Package Documentation, November 16, 2012
- Radioactive Material Shipping Personnel Qualifications, Memo to File, May 10, 2011
- WI-RSD-R-030, Part 61 Sampling Guidance, Revision 00

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- CR-PLP-2013-01451, NRC observations during review of PI Data Collection and Validation packages, April 1, 2013
- NRC Performance Indicator Technique / Data Sheet, Mitigating System Performance Indicator, Heat Removal (AFW), January 2012 through December 2012.
- NRC Performance Indicator Technique / Data Sheet, Unplanned Scrams with Complications, January 2012 through December 2012
- Palisades Mitigating System Performance Indicator Basis Document, Section 2.3, Heat Removal System, December 21, 2011

4OA2 Problem Identification and Resolution

- ANP-2547NP, Transgranular Stress Corrosion Cracking of Austenitic Stainless Steels in CRDM Applications, Revision 1
- CAP029079, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing, June 21, 2001
- C-PAL-01-02186, Root Cause Evaluation, Primary Coolant System Pressure Boundary Leakage CRD-21 Upper Housing Assembly
- CR-PLP-2012-04451, SIRW Tank leakage, June 12, 2012
- CR-PLP-2012-05623, Root Cause Evaluation Report, CRD-24 Upper Housing Leak, Revision 2

- CR-PLP-2013-01134, PCRS Condition Summary, (Criterion XVI Violation), March 15, 2013
- CR-PLP-2013-01500, PCRS Condition Summary (NRC identified Criterion V violation), April 3, 2013
- EA-C-PAL-01-2186-02, CRD Upper Housing and Nozzle Weld Susceptibility Comparison, Revision 1
- EA-EAR-2001-0373-04, Owner's Review of SI "Evaluation of Leakage from Circumferential and Axial Through-wall Cracks in Lower CRDM Housing," July 22, 2001
- EA-EAR-2001-0426-01, CRD Upper Housing Redesign, January 17, 2002
- EC 38547, Evaluation Report PLP-RPT-12-00108, Revision 0
- EN-LI-118, Root Cause Evaluation Process, Revision 18

- LPI Report A12315-LR-003, Evaluation of Inside Surface Stresses above Sub-surface Flaws at Flaw Location in CRDM #24 Upper Housing – Palisades Nuclear plant, Revision 0
- PAL-UT-12-022, T-58 SIRW Tank UT Results, July 2, 2012
- PLP-RPT-12-00108, Evaluation of 18" SIRW Tank No. T-58 F (West) Nozzle for As-Found Wall Thinning, July 4, 2012
- PLP-RPT-12-0012, Evaluation of Residual Stresses in Flaw in CRD Housing Weld Overlay – Palisades Nuclear Plant, Revision 0
- PLP-RPT-12-00121, Evaluation of Thermal Stresses at Flaw Location in CRD Upper Housing – Palisades Nuclear plant, Revision 0
- PLP-RPT-12-00123, Examination of Cracks in CRDM Housing #24, Revision 0
- PLP-RPT-12-00124
- PLP-RPT-12-00125, Leakage Calculation for CRDM Housing, Revision 0
- PLP-RPT-12-00128, Prior Evaluations of Palisades CRDM Housing, Revision 0
- PLP-RPT-13-00006, CRDM Housing at the Palisades Nuclear Plant – Recommended Future Actions, Revision 0
- PLP-RPT-13-00007, Laboratory Analysis of Leaking CRDM #24 Housing from Palisades, Revision 0
- PLP-RPT-13-00009, Summary of Technical Documents Addressing the CRDM Housing 24 cracking at the Palisades Nuclear Plant, Revision 0
- PMCR 00157732, T-58 SIRW Tank Inspection PM, November 14, 2012
- Project RP-1063, Supplier Verification Deficiency Reports, December 2001/January2002
- SOP-1B, Primary Coolant System – Cooldown, Revision 15
- T-12241-HK, Low Frequency Electromagnetic Technique of Safety Injection Refueling Water Tank, July 2, 2012
- WCAP-16000, Review of the Root Cause Evaluation for Leakage from Palisades CRD-21 Upper Housing Assembly C-PAL-01-2186, October 2003
- WI0PCS-M-06, NSSS Walkdown, Revision 3
- WPS 1149-3, Welding Procedure Specification (GTAW), Revision 3

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
ALI	Annual Limit on Intake
ALPS	Alternative Radwaste Processing System
AFW	Auxiliary Feedwater
CAP	Corrective Action Program
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
DAC	Derived Air Concentration
EC	Engineering Change
GL	Generic Letter
ID	Inside Diameter
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
LORT	Licensed Operator Requalification Training
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NDE	Non-Destructive Examination
NEI	Nuclear Energy Institute
NVLAP	National Voluntary Laboratory Accreditation Program
NRC	U.S. Nuclear Regulatory Commission
OD	Outer Diameter
PARS	Publicly Available Records System
PCS	Primary Coolant System
PI	Performance Indicator
RCR	Root Cause Report
RCT	Root Cause Team
SAT	Systems Approach to Training
SCAQ	Significant Condition Adverse to Quality
SDP	Significance Determination Process
SIRW	Safety Injection Refueling Water
SIT	Special Inspection Team
TGSCC	Transgranular Stress Corrosion Cracking
TLD	Thermoluminescent Dosimeters
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UN	United Nations
URI	Unresolved Item
UT	Ultrasonic Examination

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Sincerely,

/RA/

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

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