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

November 8, 2011

EA-11-061

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 05000255/2011004; EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Vitale:

On September 30, 2011, 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 October 12 with Mr. C. Arnone and other members of your staff.

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

Based on the results of this inspection, two NRC-identified and two self-revealed findings of very low safety significance were identified. Three of the findings involved violations of NRC requirements. Additionally, two licensee identified findings of very low safety significance are listed in the report. The inspectors screened one NRC-identified violation and determined that it warranted enforcement discretion per the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues. The remaining findings, because of their very low safety significance, and because the issues were entered into your corrective action program, are being treated as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy. If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Palisades Nuclear Plant. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Palisades Nuclear Plant.

In accordance with 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 document system (ADAMS). ADAMS is accessible from the NRC Website 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 05000255/2011004;
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:	05000255/2011004
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	July 1, 2011, to September 30, 2011
Inspectors:	J. Ellegood, Senior Resident Inspector T. Taylor, Resident Inspector J. Cassidy, Senior Health Physicist A. Dahbur, Senior Reactor Inspector A. Scarbeary, Reactor Engineer C. Zoia, Operator Licensing Examiner
Approved by:	John B. Giessner, Chief Branch 4 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

Inspection Report 05000255/2011004; 07/01/2011 - 09/30/2011; Palisades Nuclear Plant; Radiological Hazard Assessment and Exposure Controls, Other Activities

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

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance for the licensee's failure to review and update the Severe Accident Management Guidelines (SAMGs) as required by the site's procedure review process for SAMG's. Specifically, the SAMG writers' guide and site procedures required periodic or biennial reviews of the SAMGs; however, no reviews had been performed since 2005. In addition, the licensee procedures for design changes require that design changes identify impacts on SAMGs. Because the SAMGs are not required by regulations, the inspectors determined that the failure to update the SAMGs was a finding without an associated violation. The licensee has entered the condition into their corrective action program (CAP), and performed revisions, and established electronic accessibility to the SAMGs.

  The inspectors concluded that the failure to review and update the SAMGs as required by the SAMG writers' guide and licensee procedures was a performance deficiency that warranted further evaluations through the SDP. The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated December 24, 2009, because, the performance deficiency is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the objective to ensure the reliability of systems to respond to initiating events. In addition, the SAMGs are procedures used to mitigate the effects of beyond design basis accidents and, if left uncorrected, would complicate the licensee's response to a severe accident and have the potential to lead to a more significant safety concern. The inspectors concluded that the finding was not more than very low safety significance because it did not degrade any of the mitigating system functions listed in the phase 1 screen. No cross-cutting issue existed due to the age of the issue. (Section 4OA5.3)

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a finding of very low safety significance with an associated NCV of 10 CFR 50.47(b)(4) for the failure to properly implement the approved Emergency Action Level (EAL) classification scheme. Specifically, the licensee implemented the EAL classification scheme such that an Alert (one occurrence) would not be declared, as it should be, related to degraded performance of safety related equipment as a result of flooding. The licensee has entered the condition into their CAP and conducted training to implement appropriate criteria for declaration of subject EAL.

The inspectors concluded that the failure to implement a standard emergency classification scheme emergency planning drill was a performance deficiency that warranted a significance determination using the SDP. The issue was more than minor because it is associated with the Emergency Response Organization performance attribute of the Emergency Preparedness Cornerstone, and adversely affected the cornerstone objective to ensure that the capability of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency is maintained by the licensee. The issue was of very low safety significance (Green) because it met the example for a Green finding using IMC 0609 Appendix B, "Emergency Preparedness SDP" under Section 4.4 and did not meet the threshold for a greater than green finding in Appendix B since there was no loss or degradation of a Risk-Significant Planning Standard. The finding had an associated cross-cutting aspect under the area of human performance in the resources component. Specifically, the licensee did not provide adequate training of personnel. H.2(b) (Section 4OA5.2)

Cornerstone: Occupational Radiation Safety

- Green. A finding of very low safety significance and an NCV was self-revealed following the licensee's failure to control dose to workers as specified in the radiation work permit (RWP) and as required by Technical Specification (TS) 5.7.2. Specifically, inadequacies in the licensee's process for performing remote dose monitoring, resulted in workers exceeding their authorized RWP dose limits. Therefore the dose was not controlled as required by TS. The licensee has entered the condition into their corrective action program (CAP). Corrective actions included revising procedures for remote radiological job coverage for workers wearing multiple dosimeters.

  The finding was more than minor because it is addressed in Example 6.h of IMC 0612 Appendix E, "Examples of Minor Issues." Additionally, the inspectors determined that the finding was more than minor because it is associated with the program and process attribute, and affected the Occupational Radiation Safety Cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operations. This finding was assessed using IMC 0609, Attachment C for the Occupational Radiation Safety SDP and determined to be of very low safety significance (Green) because this failure did not involve as-low-as-is-reasonably-achievable (ALARA) planning or work controls; did not result in an overexposure or substantial potential for overexposure and there was not a compromised ability to assess dose. The finding was caused by vague procedural guidance. Consequently, this finding had a cross-cutting aspect in the area of human performance resources. Specifically, the licensee ensures that resources are available and adequate to maintain complete, accurate, and up to date procedures. H.2(c) (Section 2RS1.5)

- Green. A self-revealed finding of very low-safety-significance and associated NCV of TS 5.7.1, occurred when an individual entered a high radiation area without proper authorization. The individual was not knowledgeable of dose rates in the area. The licensee has entered the condition into their CAP. Corrective actions included counseling of the worker and the error was discussed with all Nuclear Plant Operators at shift turnover.

  The finding was more than minor because it is addressed in Example 6.h of IMC 0612 Appendix E, "Examples of Minor Issues." Additionally, the inspectors determined that the finding was more than minor because it is associated with the program and process attribute, and affected the Occupational Radiation Safety Cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operations. This finding was assessed using IMC 0609, Attachment C for the Occupational Radiation Safety SDP and determined to be of very low safety significance (Green) because this failure did not involve ALARA Planning or Work controls; did not result in an overexposure or substantial potential for overexposure and there was not a compromised ability to assess dose. The finding was caused by the worker that did not ask for a peer check before entering the posted high radiation area. Consequently, this finding had a cross-cutting aspect in the area of human performance work practices. Specifically, human error prevention techniques, such as self and peer checking are used. H.4(a) (Section 2RS1.7)

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 CAP. 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 at 100 percent power. On August 23, 2011, an Unusual Event (UE) was declared for an earthquake whose epicenter was in Virginia. The UE was exited approximately 4 hours later after walkdowns indicated there was no impact on the plant. On September 16, the plant commenced a reactor shutdown due to primary coolant unidentified leakage exceeding the Technical Specification (TS) limit of 1 gallon per minute. During the shutdown, the leak rate increased to greater than 10 gallons per minute, necessitating a manual reactor trip and declaration of a UE. The UE was exited approximately 5 hours later after the licensee isolated the pressurizer spray valve in containment that was leaking. The reactor was subsequently taken critical on September 19 and returned to 100 percent power on September 21. On September 25, the reactor automatically tripped when a maintenance activity inadvertently caused an electrical transient on the direct current (DC) distribution system. The reactor remained shut down for the rest of the inspection period.

1. REACTOR SAFETY

Cornerstones: 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:

- Service water during ‘B’ pump 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 (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program (CAP) with the appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On September 6 through 9, 2011, the inspectors performed a complete system alignment inspection of the critical 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 line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

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:

- diesel generator 1-2 and fuel oil day tank rooms;
- cable spreading room;
- east engineering safeguards room; and
- spent fuel pool heat exchanger 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, 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.

1R11 Licensed Operator Requalification Program (71111.11)

a. Inspection Scope

On September 21, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- 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 one quarterly maintenance effectiveness sample as defined in IP 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- Planned molluscicide treatment during potential hot weather;
- ‘C’ service water pump work after coupling failure;
- ‘B’ service water pump coupling replacements; and
- DC electrical system maintenance.

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.

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

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- reactor protection system (RPS) following spurious trip indications;
- operability of service water pumps following coupling failure; and
- operability of DC electrical system following repair from transient.

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

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

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- high pressure safety injection/ low pressure safety injection flow loop 1A square rooter replacement testing;
- diesel generator 1-2 (K-6B) testing after governor replacement;
- ‘C’ service water pump following coupling replacement;
- ‘B’ service water pump following coupling replacement;
- repair and inspection of auxiliary feedwater (AFW) check valves; and
- repair of pressurizer spray valve following failure.

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 Force Outage due to High Unidentified Leak Rate

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on September 16, 2011, and continued through September 19, 2011. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed the reactor shutdown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, startup activities, and identification and resolution of problems associated with the outage. The outage resulted from failure of packing on a pressurizer spray valve.

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

b. Findings

No findings were identified.

.2 Forced Outage Due to Loss of DC Buss

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on September 25, 2011, and continued through October 2, 2011. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed the reactor shutdown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, and control of containment activities. Since the forced outage continued through the last day of the quarter, startup activities were not included in this inspection period. Therefore, this does not constitute an inspection sample.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- TS Surveillance Procedure Daily/Weekly Operations-1 (Primary Coolant System (PCS)Leakage);
- Daily/Weekly Operations-13 test of containment airlock doors;
- AFW actuation system testing; and
- RPS trip unit testing.

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

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;

- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency was in accordance with TSs, the UFSAR, procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;
- where applicable for inservice testing activities, testing was performed in accordance with the applicable version of Section XI, 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 three routine surveillance testing samples and one reactor coolant system leak detection inspection sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

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

.1 Inspection Planning (02.01)

a. Inspection Scope

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

b. Findings

No findings were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

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

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

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

- Radiation Work Permit (RWP) 2010435; Refuel Project Shielding;
- RWP 2010468; pressurizer spray controls valves (CV) CV-1057 and CV-1059; and
- RWP 2010471 ISI Alloy 600 FAC Exams in Containment.

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

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

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

b. Findings

No findings were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

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

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

- RWP 2010435; Refuel Project Shielding;
- RWP 2010468; Pressurizer Spray Controls Valves CV-1057 and CV-1059; and
- RWP 2010471 ISI Alloy 600 FAC Exams in Containment.

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

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

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

b. Findings

No Findings were identified

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

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

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

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

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

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

b. Findings

No findings were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

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

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

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

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

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

- RWP 2010435; Refuel Project Shielding;
- RWP 2010468; Pressurizer Spray Controls Valves CV-1057 and CV-1059; and
- RWP 2010471 ISI Alloy 600 FAC Exams in Containment.

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

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

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

b. Findings

Introduction: A finding of very low safety significance and an NCV was self-revealed following the licensee's failure to control dose to workers as specified in the RWP and as required by TSs. Specifically, inadequacies in the licensee's process for performing remote dose monitoring, resulted in workers exceeding their authorized RWP dose limits.

Description: Palisades Nuclear Power Plant performed a manual bare metal inspection of the reactor vessel head during Unit 1 refueling outage 21 (1R21) and previous refueling outages. This inspection is performed using a pole with a mirror the end and a high intensity flashlight.

The workers performing the bare metal inspections were outfitted with multiple dosimeters to monitor the Effective Dose Equivalent. This process used primary and secondary dosimeters placed on the head, chest, right upper arm, left upper arm, right thigh, and left thigh. One of the secondary dosimeters also transmitted the accumulated dose and instantaneous dose rate to a remote location that was monitored by a radiation protection technician (RPT). The workers were briefed to radiological conditions for the activity. During the brief, the licensee decided that each worker would place their transmitting dosimeter on their thigh. The licensee determined that this was the location closest to the highest dose rates identified on the radiological survey. However, the RPT that performed the remote radiological job coverage was not part of the brief. In order to complete the bare metal inspection the pipe insulation needed to be removed. A group of workers removed the insulation prior to the inspection. The RPT had also provided remote radiological job coverage for workers that removed the pipe insulation. The insulation workers wore their transmitting dosimeter on the chest.

The work group assembled at the job site where face shields and radio headsets were given to the workers. The inspection started after a check of radio communications and transmitting dosimeters was performed for all workers. During the licensee's bare metal inspection an NRC inspector that observed the activity questioned the adequacy of the inspection. Specifically, the NRC questioned whether the workers were at an appropriate distance to maintain the minimum lighting as required by plant procedures and associated code requirements. The workers returned to job site to perform a closer inspection as indicated in plant procedures. At one point during the inspection, a worker thought he heard an alarm and contacted the RPT performing remote radiological job coverage. The RPT informed the worker that he did not have an alarm on the remote radiological job coverage computer screen and informed the worker that he received 170 mR of the 300 mR that was authorized for the activity. The RPT that performed remote radiological job coverage periodically informed the workers of the accumulated dose and dose rate reported by transmitting dosimeter. At the completion of the inspection, the workers exited the work area and an RPT was waiting to assist the workers removing the outer anti-contamination clothing and multiple dosimeters and identified four dosimeters that were in alarm status with the following values:

- 448.7 mR – Head;
- 453.9 mR – Chest;
- 351.3 mR – Right Upper Arm; and
- 319.8 mR – Left Upper Arm.

The licensee reported this event as a PI Occurrence in accordance Nuclear Energy Institute (NEI) 99-02 Revision 6 and determined that the cause for the additional dose was vague procedural guidance. The licensee identified several areas for improvement for monitoring multiple body locations and using effective dose equivalent. The licensee indicated that they were considering requiring transmitting dosimeters for all locations that being monitored.

Analysis: The failure to maintain radiation exposure within the values specified in the RWP is a performance deficiency, because the licensee failed to meet this requirement and the cause of this issue was reasonably within its ability to foresee and correct, and should have been prevented.

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

The inspectors reviewed the guidance in IMC 0612, Appendix E, Examples of Minor Issues, and determined that this performance deficiency is addressed in Example 6.h. and determined this to be more than minor. Additionally, the inspectors determined that the finding was more than minor because it is associated with the program and process attribute, and affected the Occupational Radiation Safety Cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation, in that the ineffective method used to remotely monitor the dose and dose rate caused the worker to receive additional dose. This finding was assessed using IMC 0609, Attachment C for the Occupational Radiation Safety SDP and determined to be of very low safety significance (Green) because this failure did not involve ALARA Planning or Work controls; did not result in an overexposure or substantial potential for overexposure and there was not a compromised ability to assess dose.

As described above, the cause for this finding dose was vague procedural guidance. Consequently, this finding had a cross-cutting aspect in the area of human performance-resources. Specifically, the licensee ensures that resources are available and adequate to maintain complete, accurate, and up to date procedures. H.2(c)

Enforcement: The licensee elected to control this entry into an area with dose rates greater than 1000 mR/hour using TS 5.7.2.d.2. Technical Specification 5.7.2.d.2 requires each group or individual entering the area shall posses a radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by RP personnel responsible for controlling personnel radiation exposure within the area, and with the means to communicate with and control every individual in the area. Contrary to the above, on October 17, 2010, the RP personnel could not effectively control exposure because the portion of the body with the highest exposure was not transmitted to the remote receiver. This was a violation. Corrective actions included revising procedures for remote radiological job coverage for workers wearing multiple dosimeters. Since the issue was of very low safety significance and has been entered in the licensee's CAP as CR-PLP-2010-5086 this violation is being treated as a Non-Cited Violation (NCV), consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2011004-01; Failure to Control Dose to Worker in Locked High Radiation Area).

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

a. Inspection Scope

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

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

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

b. Findings

No findings were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

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

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

b. Findings

Introduction: The inspectors determined a self-revealed finding of very low safety significance as well as an associated NCV of TS 5.7.1 occurred for entering a high radiation area without proper authorization by an individual that was not knowledgeable of dose rates in the area.

Description: A Nuclear Plant Operator (NPO) trainee was assigned to perform duties as Foreign Material Exclusion (FME) monitor in containment during the refueling outage. The NPO used the RWP for the activity and RP briefed the NPO for the activity. The NPO was issued an electronic dosimeter with alarm setpoints appropriate for the FME monitor assignment in a radiation area. When the FME monitor assignment was completed, the NPO decided he would go into the pressurizer shed to gain familiarity with the equipment in the room, an activity that was not part of the RP briefing or authorization. The pressurizer shed was barricaded and posted as high radiation area with dose rates that exceeded 100 mR/hour but less than 1000 mR/hour. The NPO received a dose rate alarm on the electronic dosimeter that was worn during the walkdown. The NPO immediately left the area, notified RP of the alarm and areas of the walkdown. The maximum dose rate that the worker entered was 121 mR/hour.

This NPO decided to pass through the barricade, enter the high radiation area and perform this walkdown since he believed it was acceptable as he was already in the containment building. The NPO did not ask the RPT assigned the area for permission or for a peer check before entering the pressurizer shed as he should have done in accordance with RWP 20100436 Task 1.

Analysis: The failure to comply with established radiological barriers and protective measures as specified for entry into and work within a high radiation area is a performance deficiency, because the licensee failed to meet this requirement and the cause of this issue was reasonably within its ability to foresee and correct, and should have been prevented.

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

The inspectors reviewed the guidance in IMC 0612, Appendix E, Examples of Minor Issues, and determined that this performance deficiency is addressed in Example 6.h. and determined this to be more than minor. Additionally, the inspectors determined that the finding was more than minor because it is associated with the program and process attribute and affected the Occupational Radiation Safety Cornerstone objective to ensure the adequate protection of the worker health and safety from exposure to radiation from radioactive material during routine civilian nuclear reactor operation, in that entering high radiation areas without authorization and without knowledge of dose rate in the area could lead to additional exposure. This finding was assessed using IMC 0609, Attachment C for the Occupational Radiation Safety SDP and determined to be of very low safety significance (Green) because this failure did not involve As-Low-As-Is-Reasonably-Achievable (ALARA) Planning or Work controls; did not result in an overexposure or substantial potential for overexposure and there was not a compromised ability to assess dose.

As described above, the NPO did not ask for a peer check before entering pressurizer shed an area that was barricaded and posted as a high radiation area. Consequently, this finding had a cross-cutting aspect in the area of human performance - work practices. Specifically, human error prevention techniques, such as self and peer checking were not used. H.4(a)

Enforcement: Technical Specification 5.7.1 states, in part, entry into high radiation areas shall be made only after the dose rates in area have been determined and entry personnel are knowledgeable of them. Contrary to the above, on October 15, 2010, a person entered a high radiation area and was not knowledgeable of the dose rates as required by TS 5.7.1. This was a violation. Corrective actions included counseling of the worker and the error was discussed with all NPOs at shift turnover. Since the issue was of very low safety significance and has been entered in the licensee's CAP as CR-PLP-2010-5086 this violation is being treated as a NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy (NCV 05000255/2011004-02; Unauthorized Entry to High Radiation Area)

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

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

b. Findings

No findings were identified.

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

This inspection constituted one complete sample as defined in IP 71124.02-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

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

The inspectors reviewed the site-specific trends in collective exposures (using NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities," and plant historical data) and source term (average contact dose rate with reactor coolant piping) measurements (using Electric Power

Research Institute TR-108737, “BWR Iron Control Monitoring Interim Report,” issued December 1998, and/or plant historical data, when available).

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

b. Findings

No findings were identified.

.2 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors selected the following work activities of the highest exposure significance.

- RWP 2010435; Refuel Project Shielding;
- RWP 2010468; Pressurizer Spray Controls Valves CV-1057 and CV-1059; and
- RWP 2010471 ISI Alloy 600 FAC Exams in Containment.

The inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors determined whether the licensee reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

The inspectors assessed whether the licensee’s planning identified appropriate dose mitigation features; considered alternate mitigation features; and defined reasonable dose goals. The inspectors evaluated whether the licensee’s ALARA assessment had taken into account decreased worker efficiency from use of respiratory protective devices and/or heat stress mitigation equipment (e.g., ice vests). The inspectors determined whether the licensee’s work planning considered the use of remote technologies (e.g., teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose and the use of dose reduction insights from industry operating experience and plant-specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and RWP documents.

The inspectors compared the results achieved (dose rate reductions, person-rem used) with the intended dose established in the licensee’s ALARA planning for these work activities. The inspectors compared the person-hour estimates provided by maintenance planning and other groups to the RP group with the actual work activity time requirements, and evaluated the accuracy of these time estimates. The inspectors assessed the reasons (e.g., failure to adequately plan the activity, failure to provide sufficient work controls) for any inconsistencies between intended and actual work activity doses.

The inspectors determined whether post-job reviews were conducted and if identified problems were entered into the licensee’s CAP.

b. Findings

No findings were identified.

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

a. Inspection Scope

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

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

The inspectors evaluated the licensee's method of adjusting exposure estimates, or re-planning work, when unexpected changes in scope or emergent work were encountered. The inspectors assessed whether adjustments to exposure estimates (intended dose) were based on sound RP and ALARA principles or if they were just adjusted to account for failures to control the work. The inspectors evaluated whether the frequency of these adjustments called into question the adequacy of the original ALARA planning process.

b. Findings

No findings were identified.

.4 Source Term Reduction and Control (02.04)

a. Inspection Scope

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

b. Findings

No findings were identified.

.5 Radiation Worker Performance (02.05)

a. Inspection Scope

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

b. Findings

No findings were identified.

.6 Problem Identification and Resolution (02.06)

a. Inspection Scope

The inspectors evaluated whether problems associated with ALARA planning and controls are being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee's CAP.

b. Findings

No findings were identified.

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

This inspection constituted one complete sample as defined in IP 71124.03-05.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the plant UFSAR to identify areas of the plant designed as potential airborne radiation areas and any associated ventilation systems or airborne monitoring instrumentation. Instrumentation review included continuous air monitors (continuous air monitors and particulate-iodine-noble-gas-type instruments) used to identify changing airborne radiological conditions such that actions to prevent an overexposure may be taken. The review included an overview of the respiratory protection program and a description of the types of devices used. The inspectors reviewed UFSAR, TSs, and emergency planning documents to identify location and quantity of respiratory protection devices stored for emergency use.

Inspectors reviewed the licensee's procedures for maintenance, inspection, and use of respiratory protection equipment including self-contained breathing apparatus as well as procedures for air quality maintenance.

The inspectors reviewed reported PIs to identify any related to unintended dose resulting from intakes of radioactive material.

b. Findings

No findings were identified.

.2 Engineering Controls (02.02)

a. Inspection Scope

The inspectors reviewed the licensee's use of permanent and temporary ventilation to determine whether the licensee uses ventilation systems as part of its engineering controls (in lieu of respiratory protection devices) to control airborne radioactivity. The inspectors reviewed procedural guidance for use of installed plant systems, such as containment purge, spent fuel pool ventilation, and auxiliary building ventilation, and assessed whether the systems are used, to the extent practicable, during high-risk activities (e.g., using containment purge during cavity floodup).

The inspectors selected installed ventilation systems used to mitigate the potential for airborne radioactivity, and evaluated whether the ventilation airflow capacity, flow path (including the alignment of the suction and discharges), and filter/charcoal unit efficiencies, as appropriate, were consistent with maintaining concentrations of airborne radioactivity in work areas below the concentrations of an airborne area to the extent practicable.

The inspectors selected temporary ventilation system setups (high-efficiency particulate air/charcoal negative pressure units, down draft tables, tents, metal "Kelly buildings," and other enclosures) used to support work in contaminated areas. The inspectors assessed whether the use of these systems is consistent with licensee procedural guidance and ALARA concept.

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

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

b. Findings

No findings were identified.

.3 Use of Respiratory Protection Devices (02.03)

a. Inspection Scope

For those situations where it is impractical to employ engineering controls to minimize airborne radioactivity, the inspectors assessed whether the licensee provided respiratory protective devices such that occupational doses are ALARA. The inspectors selected work activities where respiratory protection devices were used to limit the intake of radioactive materials, and assessed whether the licensee performed an evaluation concluding that further engineering controls were not practical and that the use of respirators is ALARA. The inspectors also evaluated whether the licensee had established means (such as routine bioassay) to determine if the level of protection

(protection factor) provided by the respiratory protection devices during use was at least as good as that assumed in the licensee's work controls and dose assessment.

The inspectors assessed whether respiratory protection devices used to limit the intake of radioactive materials were certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration or have been approved by the NRC per 10 CFR 20.1703(b). The inspectors selected work activities where respiratory protection devices were used. The inspectors evaluated whether the devices were used consistent with their National Institute for Occupational Safety and Health/Mine Safety and Health Administration certification or any conditions of their NRC approval.

The inspectors reviewed records of air testing for supplied-air devices and self-contained breathing apparatus bottles to assess whether the air used in these devices meets or exceeds Grade D quality. The inspectors reviewed plant breathing air supply systems to determine whether they meet the minimum pressure and airflow requirements for the devices in use.

The inspectors selected several individuals qualified to use respiratory protection devices, and assessed whether they have been deemed fit to use the devices by a physician.

Due to limited in-field observations, the inspectors reviewed training curricula for users of respiratory protection devices and requested a demonstration of device use (donning, doffing, functional checks, and device malfunction) from selected individuals.

The inspectors chose multiple respiratory protection devices staged and ready for use in the plant or stocked for issuance for use. The inspectors assessed the physical condition of the device components (mask or hood, harnesses, air lines, regulators, air bottles, etc.) and reviewed records of routine inspection for each. The inspectors selected several of the devices and reviewed records of maintenance on the vital components (e.g., pressure regulators, inhalation/exhalation valves, hose couplings). .

b. Findings

No findings were identified.

.4 Self-Contained Breathing Apparatus for Emergency Use (02.04)

a. Inspection Scope

Based on the UFSAR, TSs, and emergency operating procedure requirements, the inspectors reviewed the status and surveillance records of self-contained breathing apparatuses staged in-plant for use during emergencies. The inspectors reviewed the licensee's capability for refilling and transporting self-contained breathing apparatus air bottles to and from the control room and operations support center during emergency conditions.

The inspectors selected several individuals on control room shift crews and from designated departments currently assigned emergency duties (e.g., onsite search and rescue duties) to assess whether control room operators and other emergency response and RP personnel (assigned in-plant search and rescue duties or as required by emergency operating procedures or the emergency plan) were trained and qualified in the use of self-contained breathing apparatuses (including personal bottle change out). The inspectors evaluated whether personnel assigned to refill bottles were trained and qualified for that task.

The inspectors determined whether appropriate mask sizes and types are available for use (i.e., in-field mask size and type match what was used in fit-testing). The inspectors determined whether on-shift operators had no facial hair that would interfere with the sealing of the mask to the face and whether vision correction (e.g., glasses inserts or corrected lenses) was available as appropriate.

The inspectors reviewed the past 2 years of maintenance records for select self-contained breathing apparatus units used to support operator activities during accident conditions and designated as “ready for service” to assess whether any maintenance or repairs on any self-contained breathing apparatus unit’s vital components were performed by an individual, or individuals, certified by the manufacturer of the device to perform the work. The vital components typically are the pressure-demand air regulator and the low-pressure alarm. The inspectors reviewed the onsite maintenance procedures governing vital component work to determine any inconsistencies with the self-contained breathing apparatus manufacturer’s recommended practices. For those Self-contained breathing apparatuses designated as “ready for service,” the inspectors determined whether the required, periodic air cylinder hydrostatic testing was documented and up to date, and the retest air cylinder markings required by the U. S. Department of Transportation were in place.

b. Findings

No findings were identified.

.5 Problem Identification and Resolution (02.05)

a. Inspection Scope

The inspectors evaluated whether problems associated with the control and mitigation of in-plant airborne radioactivity were being identified by the licensee at an appropriate threshold and were properly addressed for resolution in the licensee CAP. The inspectors assessed whether the corrective actions were appropriate for a selected sample of problems involving airborne radioactivity and were appropriately documented by the licensee.

b. Findings

No findings were identified.

3. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index - Residual Heat Removal System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Residual Heat Removal System PI for the period from the Third Quarter of 2010 through the Second Quarter of 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI 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, MSPI derivation reports, and applicable surveillance documentation 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 residual heat removal system sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the reactor coolant system specific activity PI for the period from the third quarter 2010 through the second quarter 2011. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's reactor coolant system chemistry samples, TS requirements, issue reports, event reports, and NRC Integrated Inspection Reports for the period of third quarter 2010 through the second quarter 2011 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 reactor coolant system specific activity sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the occupational radiological occurrences PI for the period from the third quarter 2010 through the second quarter 2011. The inspectors used PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, to determine the accuracy of the PI data reported during those periods. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with RP staff, the scope and breadth of its data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate and accumulated dose alarms and dose reports and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one occupational exposure control effectiveness sample as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

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

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

In order to assist with the identification of repetitive equipment failures and specific human performance issues for 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.

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

.1 Unusual Event due to Seismic Event

a. Inspection Scope

The inspectors reviewed the plant's response to a seismic event that was felt onsite. On August 23, members of the licensee staff in the training building felt slight movement of the building. These staff members reported the movement to the control room. The control room staff contacted the National Earthquake Information Center and received confirmation that an earthquake centered in Virginia occurred and may have resulted in ground motion detectable at Palisades. Per the licensee's emergency plan, the combination of a seismic event felt on the plant site coupled with confirmation from the National Earthquake Information Center meets the threshold for declaration of an UE. The licensee confirmed the seismic event at 1442 EDT and declared the UE at 1452. As part of the licensee actions, the licensee reviewed plant logs and system parameters. This review did identify some motion in liquid levels in some plant tanks. The licensee did not identify any adverse affects on the plant. The licensee also performed walkdowns of the plant to verify the seismic event did not cause plant damage. The walkdowns did not identify any damage to the plant. The inspectors performed an independent walk down of the facility and did not identify any damage. The inspectors evaluated the performance of the licensee and the reporting completed by the site. The licensee exited the UE at 1825. Documents reviewed in this inspection are listed in the Attachment.

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

b. Findings

No findings were identified.

.2 Unusual Event due to High PCS leakage

a. Inspection Scope

On September 16, the plant experienced increased PCS leakage through the packing of a pressurizer spray valve. When PCS leakage exceeded 10 gallons per minute, the licensee declared a UE pursuant to their emergency plan. The inspectors responded to the control room and observed actions to shutdown the reactor and isolate the leak. The inspectors evaluated the performance of the licensee, including emergency operating procedures, the performance of mitigating systems and the reporting completed by the site. After closing isolation valves in the line for the spray valve, the leaked stopped. The licensee exited the event at 1934. The licensee's formal cause evaluation had not been completed by the end of this inspection period. The inspectors will review the circumstances surrounding the issue and the cause evaluation in the fourth quarter to determine if a performance deficiency exits. Documents reviewed in this inspection are listed in the Attachment.

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

b. Findings

No findings were identified.

.3 Reactor Trip Due to Electrical Transient

a. Inspection Scope

On September 25, 2011, two bus bars came into contact while maintenance was being performed in a DC electrical panel. The resulting short circuit caused an electrical transient on the system which caused power to be lost to several left train DC bus components. As a result of the loss of DC power, the two preferred AC electrical busses supplied by left train DC sources lost power. Sensing the loss of preferred AC power, several safety systems actuated and a reactor trip occurred. The right train DC distribution system and its associated preferred AC sources remained operable. NRC resident inspectors responded to the site to observe the stabilization of the plant. A Special Inspection Team was formed to inspect the circumstances surrounding the maintenance performed and actions taken by the licensee in response to the event. The Special Inspection Team completed their inspection activities on site and the details of the inspection are under NRC review. The results of the inspection will be publically available after the review and will be documented in Inspection Report 05000255/2011014.

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) Unresolved Item 05000255/2007005-01; "Incorrect Instructions to Operators in Appendix R, Post-Fire Alternate Safe Shutdown Procedure"

a. Inspection Scope

During the 2007 triennial fire protection inspection, the inspectors identified a finding concerning incorrect instructions to operators in the licensee's Post-Fire Safe shutdown (SSD) procedure that did not satisfy 10 CFR Part 50, Appendix R, Section III.L. Since the finding was indirectly related to a circuit analysis issue and the licensee was in transition to National Fire Protection Association (NFPA) 805, the finding was considered a URI pending the licensee's completion of a risk-assessment evaluation to determine the risk significance in accordance with the established NRC Enforcement Discretion regarding plants in transition to NFPA 805.

b. Findings

Incorrect Instructions to Operators in Appendix R, Post-Fire Alternate Safe shutdown Procedure

Introduction: The following finding that affected 10 CFR 50.48 was identified by the NRC and was a violation of NRC requirements. This finding has been screened and determined to warrant enforcement discretion per Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues.

The inspectors identified a finding of very low safety significance and associated violation of 10 CFR 50.48(b)2 for the licensee's failure to have procedures to implement its alternate shutdown capability, as specified in Section III.L3 of 10 CFR Part 50, Appendix R. Specifically, from December 6, 1995 to May 10, 2007, the safe shutdown analysis reports and the associated off normal alternate shutdown procedures were inconsistent in that, they specified an incorrect electrical power supply distribution cabinet number (specifying the location of a 125 Vdc breaker) for power feed to 13 post-fire safe shutdown control valves (CVs).

Description: During the 2007 triennial fire protection inspection, the inspectors identified several discrepancies in the Compliance Strategies provided in Safe Shutdown Analysis Report EA-APR-95-007, for safe shutdown valves CV-1318, Service Water Header Isolation Valve, and CV-1359, Service Water to Non-Critical Loads. Specifically, for CV-1318, the Safe Shutdown Analysis Report stated that a fire in FA 9 "Intake Structure," assumed to damage cables to CV-1318 and may cause the valve to fail open or spuriously close. The Compliance Strategy in the analysis for safe shutdown valve CV-1318 stated, "de-energize CV-1318 at ED-11-2, Breaker 72-129 in Room 224 (cable spreading room) in order to fail the valve open." A similar compliance strategy was stated for CV-1359 to fail the valve closed. The inspectors reviewed related electrical single line, schematic, and logic diagrams and determined that references to Breaker 72-129 at 125Vdc electrical distribution cabinet ED-11-2 were incorrect for safe shutdown Valves CV-1318 and CV-1359 and should have been Breaker 72-229 at

D-21-2. Similar discrepancies were noted with 11 additional safe shutdown CV circuits fed from the same Breaker 72-229, located in Cabinet D-21-2.

During the inspection, the inspectors also reviewed Attachments 3 and 5 of Procedure ONP - 25.2, “Alternate Safe Shutdown Procedure,” dated January 16, 2007, Revision 22. This procedure was used for alternate safe shutdown and to mitigate spurious operation of electrical components. The inspectors identified that Procedure ONP-25.2, Attachment 3, “Manual Actions to Mitigate the Spurious Operation of Air-Operated Valves,” provided an incorrect electrical distribution panel number as the location of the breaker for 13 valves. Specifically, Attachment 3, specified 125Vdc distribution Panel Number D21-1, as the location of Breaker 72-229 instead of the correct Panel Number D21-2. The procedure also referenced to the incorrect distribution panel number for CV-1318 and CV-1359.

In order to evaluate the significance of the procedure error, the inspectors observed a simulated operation by a former operator familiar with the subject operations procedure. The overall intent was to observe the simulated use of discrepant Procedure ONP-25.2 and determine the feasibility and timeliness of the operator actions at the alternate shutdown location of the DC distribution cabinets. The inspectors concluded, based on this observation, that in the event of a control room or intake structure fire, the operator would be challenged and potentially unsuccessful in completing the required procedure instructions/steps within the assumed times for completion of the activity. For example, the operator action required to de-energize and close CV-1359, by opening breaker 72-229 had a time constraint of 3.43 minutes from time of fire/spurious valve actuation in order to isolate the non-essential service water header and prevent overheating the diesel generator. Based on the observations, the inspectors could not conclude that the action would have been successfully completed within the specified time constraint due to the incorrect procedure.

Upon discovery, the licensee placed this issue into their CAP as AR 01087847, AR 01088311, and AR 01088327. The licensee performed an extensive extent of condition review per AR 01087847-1 for this finding and identified a number of additional similar discrepancies in Procedures ONP-25.1, ONP-25.2, several fire protection analyses, and Appendix R drawings. Consequently, after the inspection, the licensee revised the affected procedure, the safe shutdown analysis and corrected all identified discrepancies.

<u>Analysis</u>: The inspectors determined that the licensee’s failure to have correct instruction in the alternate safe shutdown procedure did not meet the provisions of 10 CFR Part 50, Appendix R, Section III.L.3, and was a performance deficiency. Specifically, the alternate safe shutdown off normal Procedure ONP-25.2 was inadequate in that it specified an incorrect electrical power supply distribution cabinet number as a feed for 13 post-fire safe shutdown CVs including CV-1318 and CV-1359. The finding was determined to be more than minor because it was associated with the Mitigating System Cornerstone attribute of Protection Against External Factors (Fire) and affected the cornerstone objective of ensuring the availability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Additonally, the finding involved the attribute of procedure quality for not providing adequate instructions in the safe shutdown procedure, which could have adversely impacted the operator’s ability to promptly take appropriate actions and could have complicated safe shutdown in the event of a fire.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase I – Initial Screening and Characterization of Findings," Tables 3b. The inspectors determined the finding degraded the fire protection defense-in-depth strategies. Therefore, screening under IMC 0609, Appendix F, "Fire Protection Significance Determination Process," was required. The inspectors assigned a low degradation for the finding per Table A.2-3 "Guidance for Ranking an Observed SSD Degradation Finding," of IMC 0609, Appendix F, Attachment 2 because the finding involved procedural deficiencies that could be compensated by operator experience and other manual actions. The off normal Procedure ONP-25.2 included steps on every odd page of the procedure to monitor the diesel generator jacket water temperature and to trip the diesel in the event of an alarm initiated due to high temperature. These actions will provide more time for recovery actions to establish necessary cooling water to the diesel in the event of a spurious operation of either CV-1318 or CV-1359. Based on the low degradation of this finding, the inspectors determined the issue was of very low safety significance (Green).

Per the Enforcement Guidance Memoranda on Fire Protection, dated April 16, 2010, cross-cutting aspects are not addressed for findings which warrant enforcement discretion per the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues.

Enforcement: Title 10 CFR 50.48(b)(2) requires, in part, that all nuclear power plants licensed to operate prior to January 1, 1979, must satisfy the applicable requirements of Appendix R to this part, including specifically the requirements of Sections III.G, III.J, and III.O. Compliance with 10 CFR Part 50, Appendix R, Section III.L is considered necessary in order to satisfy the requirements of 10 CFR Part 50, Appendix R, Section III.G. Section III.L of 10 CFR Part 50, Appendix R, specifies measures to implement alternative and dedicated shutdown capability required by Section III.G.3 of 10 CFR Part 50, Appendix R. Section III.L.3 of 10 CFR Part 50, Appendix R, states, in part, that alternative shutdown capability shall be independent of the specific fire area and that procedures shall be in effect to implement this capability. Off Normal Procedure ONP-25.2 is used to safely shutdown the plant in event of a fire in any of the alternate shutdown areas.

Contrary to the above, from 1995 to May 2007, the licensee failed to provide adequate procedural instructions to implement an alternative shutdown capability. Specifically, Attachments 3 and 5 of ONP-25.2 failed to provide the correct electrical power supply distribution cabinet number for the power feeds to 13 post-fire safe shutdown CVs. The procedure instructed the operators to open Circuit Breaker 72-229 at DC Distribution Panel D21-1 to de-energize the affected CVs in order to mitigate potential spurious operations of these valves. However, Circuit Breaker 72-229 was actually located at DC Distribution Panel D21-2. In May 2007, the licensee revised the affected procedure including ONP-25.2 based on the inspector finding and corrected all identified discrepancies.

The licensee is in transition to NFPA 805; and therefore, the NRC-identified violation was evaluated in accordance with the criteria established by Section A of the NRC's Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues (10 CFR 50.48) for a licensee in NFPA 805 transition. The inspectors determined that for this violation: (1) the licensee would have identified the violation during the scheduled transition to 10 CFR 50.48(c); (2) the licensee had established adequate compensatory measures within a reasonable time frame following identification and has corrected the violation; (3) the violation was not likely to have been previously identified by routine licensee efforts; and (4) the violation was not willful. The finding also met additional criteria established in Section 12.01.b of IMC 0305. In addition, in order for the NRC to consider granting enforcement discretion the violation must not be associated with a finding of high safety significance (i.e., Red). Therefore, the licensee performed a risk-evaluation and determined that this issue was not associated with a finding of high safety significance. In addition, the licensee entered this issue into their CAP and revised the procedure to reflect the correct panel number. As a result, the inspectors concluded that the violation met all criteria established by Section A and the NRC was exercising enforcement discretion to not cite this violation in accordance with the Interim Enforcement Policy Regarding Enforcement Discretion for Certain Fire Protection Issues.

The inspectors' review of this issue was considered to be a part of the original inspection effort, and as such, did not constitute any additional inspection samples.

.2 (Closed) Unresolved Item 05000255/2011003-05; "Application of EAL Scheme during Drill"

a. Inspection Scope

In Report 05000255/2011003, the inspectors documented an unresolved item regarding a drill that occurred on February 23, 2011. Specifically, the licensee declared a UE during the drill following reports from an Auxiliary Operator that flooding in the AFW pump room had resulted in 3 feet of water in the room. The inspectors concluded that an Alert should have been declared. The inspectors discussed the issue with regional and Nuclear Safety and Incident Response experts and agreed the licensee did not correctly implement the EAL classification scheme.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance with an associated NCV of 10 CFR 50.47(b)(4) for the failure to properly implement the approved EAL classification scheme. Specifically, the licensee implemented the EAL classification scheme such that one Alert would not be declared.

Description: On February 23, 2011, the inspectors observed licensee performance during a routine integrated emergency planning drill. The scenario started with a fire water header rupture in the AFW pump room with attendant alarms in the fire suppression system. The licensee structured the scenario to allow either an UE or Alert to be declared based on the whether or not the shift manager felt there was a safety issue based on the controller-provided information. In the scenario, a drill controller estimated the water level in the room using elapsed time from the start of the flooding. In this case, the drill controller informed the auxiliary operator that they saw approximately 3 feet of water in the room. The room contains two of the three AFW pumps onsite: the turbine driven AFW pump and an electric-motor driven AFW pump. The electric motor is an open, drip-proof design and with 3 feet of water in the room, the motor of the electric AFW pump would be approximately two-thirds submerged with water having entered the lower vents. The shift manager classified the event as a UE. Per the EAL scheme, a UE in this category is described as uncontrolled flooding that has the potential to affect safety related equipment needed for the current operating mode. An Alert is described as uncontrolled flooding that results in degraded safety system performance as indicated in the control room or that creates industrial safety hazards (e.g. electric shock) that preclude access necessary to operate or monitor safety equipment. The inspectors concluded that a partially submerged motor that is not designed for submergence would degrade safety system performance. In discussions with the licensee, many members of the licensee staff believed that "as indicated in the control room" was confined to parameters monitored in the control room. In this case, the licensee believed that if no ground existed, there would be no indication of degradation of the AFW system. When the inspectors inquired about the absence of a ground indication, the licensee stated no ground would be received until the AFW motor breaker was closed. Therefore, the inspectors concluded that lack of a ground alarm could not be relied upon to demonstrate that the submerged motor was not impacted by the flooding. The EAL basis states that "the inability to operate or monitor safety related equipment represents a potential for substantial degradation of the level of safety of the plant." Given the detrimental effects of submerging the electric motor, the unknown impact of partially submerging the turbine-driven AFW pump, and the projected difficulty personnel would have in accessing the room, the inspectors determined the appropriate call was an Alert. The EAL Basis Document also states that the escalation to an Alert is based, in part, on the event causing visible damage and/or degraded system response and is intended to discriminate against lesser events. It also states that for the Alert EAL, no attempt is made to assess the actual magnitude of the damage and that the significance is not that a particular system was damaged, but that the event was of sufficient magnitude to cause the degradation. Given the conditions of the room, it is reasonable to assume degradation of the safety related equipment was present; hence an Alert was the appropriate call. By design, the drill scenario progressed via a path unrelated to the AFW room flooding and the drill participants did not have time to attempt AFW room entry or further assessment of the condition of systems, structures and components in the AFW room.

The licensee performed an apparent cause evaluation on the issue. In the apparent cause evaluation, the licensee concluded that neither training nor the EAL basis documentation provided adequate guidance for classifying the EAL. Although the licensee did not discuss the inspector's concerns in the critique, the dominant concern was the misconception on the part of the Emergency Preparedness organization and qualified watchstanders as to classification of the scenario as described. Therefore, the inspectors considered the performance deficiency to be the result of a problem in the EAL implementation process vice a critique issue. Subsequent discussions with members of the license's staff indicated inconsistencies in criteria that would be used to classify this event.

Analysis: The inspectors concluded that the failure to implement a standard emergency classification scheme emergency planning drill was a performance deficiency that warranted a significance determination using the SDP. The issue was more than minor because it impacted the Emergency Response Organization performance attribute of the Emergency Preparedness Cornerstone, and adversely affected the objective to ensure that the capability of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency is maintained by the licensee. Specifically, misconceptions on the appropriateness of an Alert vice UE given the information in the scenario resulted in improper implementation of the EAL scheme. The issue was of very low safety significance (Green) because it met the example for a

Green finding using IMC 0609 Appendix B, “Emergency Preparedness Significance Determination Process” under Section 4.4. Specifically, the EAL classification process would not declare an Alert that should be declared. In addition, the issue screens as green since no greater than green criteria was met in Section 4.4 since there was no loss or degradation of a Risk-Significant Planning Standard and was related to one process failure to declare an Alert. The finding had an associated cross-cutting aspect under the area of human performance in the resources component. Specifically, the licensee did not provide adequate training of personnel. (H.2(b)).

Enforcement: 10 CFR 50.47(b)(4) states, in part, that “a standard emergency classification scheme and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee…” Contrary to this, on February 23, 2011 during a drill, the licensee improperly implemented the approved classification and action level scheme by declaring a UE instead of an Alert. Further investigation by the licensee revealed that training and basis documents lacked sufficient guidance to result in consistent classification of the drill scenario. The licensee generated CR-PLP-2011-01850 within their CAP to address the issue. Because this violation was of very low safety significance and it was entered into the licensee’s CAP, this violation is being treated as an NCV, consistent with the Enforcement Policy: NCV 05000255/2011004-03, Failure to Implement the Approved Emergency Classification Scheme.

.3 (Closed) Unresolved Item 05000255/2011003-09; “Failure to Update Severe Accident Management Guidelines”

During review of the SAMGs pursuant to TI 2515/184, the inspectors opened an unresolved item because the licensee had not updated SAMGs since 2005. The inspectors noted that the SAMG writers’ guide required a biennial review of the SAMGs. The licensee adopted and approved the SAMG’s writers guide as the document governing the SAMGs. In addition, the licensee’s design change process requires reviews and updates of SAMGs as part of the design process. The inspectors noted that plant modifications as a result of GSI-191 potentially impacted the SAMGs but the SAMGs had not been updated to reflect the modifications. Based on review of the condition and additional guidance, the inspectors concluded that the failure to update the SAMGs was a Finding.

Introduction: The inspectors identified a finding of very low safety significance for the licensee’s failure to review and update the Severe Accident Management Guidelines (SAMGs) as required by the site’s procedure review process. The licensee had not performed periodic reviews nor had the licensee revised the SAMGs to reflect changes to the facility.

Discussion: During review of the SAMGs, the inspectors noted that although the SAMG Writers’ Guide requires the licensee to perform periodic or biennial reviews of the SAMGs, no reviews had taken place since 2005. In addition, the licensee’s Engineering Change Process requires the licensee to review SAMGs for impact as part of the design review process. The inspectors reviewed EN-DC-115, Engineering Change Process, which includes checklists guiding designers to change SAMGs, if needed. The inspectors also reviewed superseded procedures that were in effect during some periods when the GSI-191 modifications were being developed. Procedures FP-E-MOD-04, Regulatory commitments, FP-E-MOD-06, Modification Plant Impacts, FP-E-MOD-05, Modification Plan Impacts all require that engineers indentify documents that would be affected by a modification. In reviewing the licensee's assessment of SAMG adequacy, the inspectors noted that the licensee self-identified that hydrogen recombiners had not been removed from the SAMGs. However the licensee did not identify any additional changes to SAMGs that were needed. The inspectors reviewed the changes made to the sump as part of GSI-191 and concluded that the licensee had not revised the SAMGs to account for the GSI-191 modifications. For example, these modifications removed two screens internal to the sump that were associated with discreet trains of Safety Injection. GSI-191 replaced the screens with a much larger screen array that is not associated with a discreet SI train. However, the SAMGs still refer to securing one train of Safety Injection to eliminate flow through the other train. This change could affect the SAMG strategy. The inspectors also noted that the SAMGs still refer to the use of tri-sodium phosphate as a neutralizing agent in containment even though the GSI-191 mod had replaced the TSP with potassium tetraborate. The licensee wrote CR-PLP-2011-02515 and CR-PLP-05631 to address this issue. In addition, the licensee now controls the SAMGs using the same procedures and computer infrastructure as other site procedures.

Analysis: The inspectors concluded that the failure to review and update the SAMGs as required by the SAMG writers' guide and licensee procedures was a performance deficiency that warranted further evaluations through the SDP. The finding was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated December 24, 2009, because, the performance deficiency is associated with the procedure quality attribute of the Mitigating Systems Cornerstone and adversely affected the objective to ensure the reliability of systems to respond to initiating events. In addition, the SAMGs are procedures used to mitigate the effects of a beyond design basis accidents and, if left uncorrected, would complicate the licensee's response to a severe accident and have the potential to lead to a more significant safety concern. The inspectors evaluated the finding using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the Mitigating Systems Cornerstone, dated January 10, 2008. The inspectors answered "No" to the Mitigating Systems questions and screened the finding as having very low safety significance (Green). Since these issues occurred several years ago and are not indicative of current performance, no cross-cutting aspect was assigned.

Enforcement: This finding does not involve an enforcement action because no regulatory requirement violation was identified. Because this finding does not involve a violation and has very low safety significance, it is identified as FIN-05000255/2011004-04, Failure to Maintain SAMGs. The licensee entered the finding into the CAP as PLP-2011-05631.

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 12, the inspectors presented the inspection results to D. Hamilton and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The results of the review of Unresolved Item 05000255/2007005-01 concerning the discrepancies identified in the Post-Fire SSD analysis and shutdown Procedure ONP-25.2 for incorrect electrical distribution panel number for several valves were discussed with Mr. R. Bloomfield on August 23, 2011.
- The results of the inspections of the ALARA and airborne radioactivity mitigation programs with the Site Vice President, Mr. T. Vitale, and other members of your staff, on August 5, 2011.
- The results of the radiological hazard assessment program and occupational radiation safety PI verification inspections with the Site Vice President, Mr. T. Vitale, and other members of your staff, on September 2, 2011.

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

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) or Severity Level IV were 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.

- TS 5.4.1 requires written procedures be established, implemented, and maintained for procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A. Section 7.e.4 of this document recommends procedures for contamination control and the licensee established EN-RP-122, “Alpha Monitoring” Revision 5 to address this requirement. Contrary to the above, on March 22, 2011, the licensee failed to implement the requirements of EN-RP-122 for controlling work in posted Level 3 Alpha contamination area. This issue was documented in the licensee’s CAP as CR-PLP-2011-001425. Corrective actions included suspending qualifications for individuals involved, created additional training for the RP department and calculating dose for the radiation workers affected.

  The failure to control work in posted Level 3 Alpha contamination area is a performance deficiency as defined in IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening.” The inspectors determined that the finding was more than minor because if left uncorrected this could have the potential to lead to a more significant safety concern. This finding was assessed using IMC 0609, Attachment C for the Occupational Radiation Safety SDP and determined to be of very low safety significance because this failure did not involve ALARA Planning or Work controls; did not result in an overexposure or substantial potential for overexposure and there was not a compromised ability to assess dose.

- TS 5.7.1 requires for areas with high radiation areas with dose rates not exceeding 1 R/hr dose rates each entry way to such area shall be barricaded and conspicuously posted as a high radiation area. Contrary to the above, on April 28, 2011, dose rates of 220 mR/hr were identified and the area was not posted or controlled as a high radiation area. This condition was identified by a RPT performing a routine radiological survey on July 27, 2011. This issue was documented in the licensee's CAP asCR-PLP-2011-03703. Corrective actions included posting and controlling access to the area as required and evaluation changes to plant supply chain procedures associated with the receipt and traceability of radioactive material.

The failure to post and control access to a high radiation area is a performance deficiency as defined in IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening." The inspectors determined that the finding was more than minor because it is addressed in Example 6.g. in IMC 0612 Appendix E "Examples of Minor Issues." This finding was assessed using IMC 0609, Attachment C for the Occupational Radiation Safety SDP and determined to be of very low safety significance because this failure did not involve ALARA Planning or Work controls; did not result in an overexposure or substantial potential for overexposure and there was not a compromised ability to assess dose.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Bloomfield, Acting Engineering Program and Components Manager
B. Dotson, Entergy/Licensing Technical Specialist
C. Sherman, Radiation Protection Manager
T. Swiecicki, Programs Engineer
S. Weimer, Fire Protection Engineer
D. Villicana, ALARA Supervisor

Nuclear Regulatory Commission

John B. Giessner, Chief, Reactor Projects Branch 4

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000255/2011004-01	NCV	Failure to Control Dose to Worker in Locked High Radiation Area. (2RS1)
05000255/2011004-02	NCV	Unauthorized Entry to High Radiation Area. (2RS1)
05000255/2011004-03	NCV	Failure to Implement the Approved Emergency Classification Scheme (4OA5)
05000255/2011004-04	FIN	Failure to Maintain SAMGs (4OA5)

Closed

05000255/2011004-01	NCV	Failure to Control Dose to Worker in Locked High Radiation Area. (Section 2RS1)
05000255/2011004-02	NCV	Unauthorized Entry to High Radiation Area. (Section 2RS1)
05000255/2011004-03	NCV	Failure to Implement the Approved Emergency Classification Scheme (4OA5)
05000255/2011004-04	FIN	Failure to Maintain SAMGs (4OA5)
05000255/2007005-01	URI	Incorrect Instructions to Operators in Appendix R, Post-Fire Alternate Safe Shutdown Procedure
05000255/2011003-05	URI	Application of EAL Scheme During Drill (4OA5)
05000255/2011003-09	URI	Failure to Update Severe Accident Management Guidelines (4OA5)

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.

<u>1R04 Equipment Alignment</u>

- CR-PLP-2011-03207, Health Physics Performing Routine Weekly Surveys Identified Service Water Leakage from Lagging on Downstream Side of CV-0824
- CR-PLP-2011-04212, Service Water Leakage Downstream of CV-0824 has Exceeded Trigger Point #1 of ODMI
- CR-PLP-2011-04430, MV-SW242 has a Severity Level 2 Packing Leak
- CR-PLP-2011-04431, MV-FP105, Electric Fire Pump P-9A Discharge to Test Manifold, has Severity Level 2 Packing Leak
- Design Basis Document 1.02, Service Water System, Revision 8
- M-208, Sheet 1, P&ID Non-Critical Service Water, Revision 81
- M-208, Sheet 1A, P&ID Service Water System, Revision 53
- M-208, Sheet 1B, P&ID Service Water System, Revision 34
- M-208, Sheet A, System Diagram of Service Water System, Revision 19
- M-213, P&ID Service Water Screen Structure and Chlorinator, Revision 83
- SOP-15.1, Service Water System Checklist – Critical, Revision 51
- SOP-15.2, Service Water System Checklist – Noncritical, Revision 51
- SOP-15.3, Service Water System Checklist – SG Feed and Heater Drain Pumps, Revision 51
- SOP-15.4, Service Water System Checklist – Turbine Generator, Revision 51
- USAR Chapter 9, Section 9.1, Service Water Systems, Revision 27

<u>1R05 Fire Protection</u>

- CR-PLP-2011-04450, Fire Extinguisher Mounting Bracket Ripped Out of the Wall in East ESG room
- FPIP-7, Palisades Nuclear Plant Fire Protection Implementing Procedure, Revision 20
- Palisades Cable Spreading Room Pre-Fire Plan (Fire Area 2), Revision 0
- Palisades Diesel Generator 1-2 and Fuel Oil Day Tank Room Pre-Fire Plan (Fire Areas 6 & 8), Revision 0
- Palisades East Engineered Safeguards Room Pre-Fire Plan (Fire Area 10), Revision 0
- Palisades Plant Fire Hazards Analysis Report, Revision 7
- Palisades Spent Fuel Pool Heat Exchanger Room Pre-Fire Plan (Fire Area 13G), Revision 0

<u>1R11 Licensed Operator Requalification Program</u>

- EOP-9, Functional Recovery Procedure, Revision 21

<u>1R12 Maintenance Effectiveness</u>

- CR-PLP-2007-03423, C-2A, C-2B, and C-2C placed in a(1) status, August 23, 2007
- CR-PLP-2010-00524, C-2C would not start after being placed back into service, February 5, 2010
- CR-PLP-2011-03520, C-2C tripped on low oil pressure, July 17, 2011
- CR-PLP-2011-03546, C-2A was placed in hand and 52-1106 did not close, July 18, 2011

- CR-PLP-2011-04090, Portion of petcock valve cracked and broke, August 18, 2011
- CR-PLP-2011-04339, Adjusted oil pressure on C-2C per SOP-19, September 1, 2011
- DBD-1.05, Compressed Air Systems, Revision 4
- EM-32-06, Compressed Air Monitoring Program, Revision 1
- EN-DC-206, Maintenance Rule a(1) Process, Revision 1
- Instrument air system health reports, 2010-2011

1R13 Maintenance Risk Assessments and Emergent Work Control

- Admin 4.02, Control of Equipment, Revision 59
- CR-PLP-2011-04801, Received various alarms while restoring breaker 72-123, September 23, 2011
- ENS-EP-302, Severe Weather Response, Revision 11
- EN-WM-105, Planning, Revision 9
- EOOS quantitative daily risk reports
- GOP-14, Attachment 3, Shutdown Cooling Equipment Availability, September 26, 2011
- GOP-14, Shutdown Cooling Operations, Revision 43
- ONP-12, Acts of Nature, Revision 28
- P-7B and P-7C work schedules
- SWSO-4, Molluscicide Treatment of Service Water and Fire Protection Systems, Revision 16
- WD 950 E-8 sheet 2, Single Line Meter and Relay Diagram , 125vdc, Revision 48
- WO 291210, DC Bus ED-11-2

1R15 Operability Evaluations

- C-PAL-01-965, RPS channel A tell-tale lit spuriously, March 28, 2001
- C-PAL-95-0040, RPS channel A tell-tale lights spuriously lit, January 11, 1995
- CR-PLP-2011-03761, RPS Loss of load trip tell-tales illuminated unexpectedly, August 1, 2011
- CR-PLP-2011-03902, Service water pump P-7C failed unexpectedly, August 9, 2011
- CR-PLP-2011-03961, Potential extent of condition for P-7A and P-7B, August 11, 2011
- CR-PLP-2011-04835, Coordination of breaker 72-01 with other breakers in the DC system not considered, September 26, 2011
- DBD 2.05, Reactor Protective System, Safety Injection Signal, Anticipated Transient Without Scram, Revision 6
- Drawing E-8, Sheet 1, 125VDC and 120V Instrument and Preferred AC System, Revision 57
- Drawing VEN-M1-Q Sheet 4012, Auxiliary trip module schematic, Revision 0
- FSAR Section 14.7, Decreased Reactor Coolant Flow, Revision 26
- IEEE Standard 450-1995, Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications
- Operator logs, September 26, 2011
- QI-3, Reactor Protection System Logic Tests, Revision 5
- WO 285752, Test TA-0305A to verify proper operation
- WO 291210, DC Bus ED-11-2

1R19 Post-Maintenance Testing

- ASME OM Code, Section ISTC
- CR-PLP-2011-04552, Task added to perform non-intrusive test on CK-FW741, September 13, 2011
- CR-PLP-2011-04679, CV-1057 did not fully stroke, September 19, 2011
- EM-09-18, Check Valve Condition Monitoring and Inservice Testing Program, Revision 7
- EN-WM-107, Post Maintenance Testing, Revision 3

- Instrument Calibration Sheet for FM-0307, Bench Test, June 16, 2011
- Instrument Calibration Sheet for FM-0308, Bench Test, July 6, 2011
- MSM-M-57, Universal Diagnostic System Operating Procedure, Revision 9
- PCS-M-8, Repairing Pressurizer Spray Valves, Revision 18
- QO-14, Inservice Test Procedure-Service Water Pumps, Revision 34
- QO-21, Inservice Test Procedure, Auxiliary Feedwater Pumps, Revision 15
- SOP-22, Revision 50, Emergency Diesel Generators
- Tech Spec Surveillance Procedure MO-7A-2, Revision 73, Emergency Diesel Generator 1-2
- Tech Spec Surveillance Procedure RE-138, Revision 8, Bus 1D UV and Time Delay Relays
- WI-SWS-M-04, Service Water Pumps P-7B and P-7C Removal, Inspection, and Reinstallation, Revision 4
- WO 235764, CK-FW741, Repair leak at hinge pin cover
- WO 289180, P-7B, PMT after recheck of impeller setting
- WO 290401, CV-1057: Repack valve
- WO00238656, Replace EGA Governor with Spare
- WO00241446, Replace Speed Switch
- WO00250106, Replace Governor Servo Booster
- WO00258011, RE-132 Diesel Generator 1-2 Load Reject
- WO51633937, Replace Governor on Diesel Generator 1-2
- WO52224733, HPSI Flow Loop 1A Square Rooter Replacement
- WO52224738, LPSI Flow Loop 1A Square Rooter Replacement

1R20 Outage Activities

- DBD-4.02, 125VDC System (Safety-Related), Revision 9
- EC31861, Sodium Tetraborate Wetting,
- GOP-14, Shutdown Cooling Operations, Revision 43
- GOP-2, Mode 5 to Mode 3 >/= 525 degrees F, Revision 32
- GOP-4, Mode 2 to Mode 1, Revision 22
- GOP-8, Power reduction and Plant Shutdown to Mode 2 or Mode 3 >= 525oF, Revision 28,
- GOP-9, Mode 3 >/= 525 degrees F to Mode 4 or Mode 5, Revision 31
- PO-2, PCS Heatup/Cooldown Operations, Revision 4
- Proc. 4.08, Post Event Review Requirements, September 17, 2011
- SOP-1A, Primary Coolant System, Revision 16
- SOP-1B, Primary Coolant System-Cooldown, Revision 11
- SOP-1C, Primary Coolant System-Heatup, Revision 10
- SOP-3, Safety Injection and Shutdown Cooling System, Revision 80
- SOP-6, Reactor Control System, Revision 32
- SOP-8, Main Turbine and Generating System, Revision 84

1R22 Surveillance Testing

- Administrative Procedure 4.19, PCS Leak Rate Determination
- Calculated Quench Tank T-73 In Leakage Rate graph, October 29, 2010 to August 30, 2011
- CIS-M-6, Personnel Air Lock Seal Contact Adjustment, Revision 0
- CR-PLP-2011-00409, Unidentified PCS Leakage at 0.143 gpm on January 26, 2011
- CR-PLP-2011-04618, NRC Identified Manual PCS Leakrate Procedural Shortfalls
- Daily/Weekly Operations-13, Local Leak Rate Tests for Inner and Outer Personnel Air Lock Door Seals, Revision 22
- Drawing M1-Q Sheet 113, Reactor Protective System Functional Diagram, Revision F
- Drawing M1-Q Sheet 114, Block Diagram, Reactor Protective System, Revision 13

- FSAR Chapter 5, Containment Structure Testing, Revision 29
- Leak Rate Calculator, Version 1.05 (Excel Spreadsheet Application)
- PCS Leakrate Snap-Shot screen print, September 14, 2011
- Procedure Daily/Weekly Operations-1, Attachment 13, Revision 84, Subcooled Specific Volumes
- Procedure Daily/Weekly Operations-1, Revision 94, Tech Spec Surveillance for PCS Leakage
- QI-39, Auxiliary Feedwater Actuation System Logic Test, Revision 4
- QI-9, Reactor Protective Trip Units, Revision 9
- Selected Operator logs, January 26, 2011 through January 27, 2011
- T-73 In-Leakage v. PCS ULR graph from J:\OPS\Databases and Spreadsheets
- WO 284708, Excessive leakage inner door (OPS return to service)

2RS1 Radiological Hazard Assessment and Exposure Controls

- CR-PLP-2010-05186, Rx Head Bare Metal Inspection Dose Alarm, October 17, 2010
- CR-PLP-2010-05086, Radiation Worker Entered Unauthorized/Un-briefed Areas, October 15, 2010
- CR-PLP-2009-04810, Previous NRC Finding Unauthorized Entry into High Radiation Area, October 15, 2009
- WI-RSD-R-010, Instructions for Work On Contaminated Equipment in Radioactive Material Areas External to the Primary Auxiliary Building Radiologically Controlled Area, Revision 8
- SR-12, Sealed Source Leak Test, Revision 9
- EN-RP-141-01, Job Coverage Using Remote Monitoring Technology, Revision 3
- EN-RP-203, Dose Assessment, Revision 4
- EN-RP-204, Special Monitoring Requirements, Revision 4
- WI-RSD-A-023, Use of Extremity or Multi-Badging Dosimetry, Revision 0
- EN-RP-151, Radiological Diving, Revision 2
- EN-RP-121, Radioactive Material Control, Revision 6
- WI-RSD-R-025, Control and Storage of Radioactive Tools and Equipment, Revision 9
- EN-RP-101, Access Control for Radiologically Controlled Areas, Revision 6
- EN-RP-108, Radiation Protection Posting, Revision 10
- CR-PLP-2010-05060, Refuel Machine Malfunctioned, October 14, 2010
- CR-PLP-2011-03703, Un-posted High Radiation Area, July 27, 2011
- Work Order 52323127 01, SR-12 Sealed Source Leak Test, August 4, 2011
- Work Order 52302733 01, M-8 Plant Heating Boiler, August 31, 2011

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

- CR-PLP-2010-03024, Long Term Excessive Collective Radiation Exposure at Palisades Nuclear Plant, September 1, 2010
- CR-PLP-2010-04682, ALARA Techs and ALARA Analyst Pulled to Support RP Operations, October 8, 2010
- CR-PLP-2010-06206, Palisades Failed to Meet Outage Dose Goal, February 11, 2011
- CR-PLP-2011-001425, Lapel Air Sample Results Indicate 24 DAC Alpha, March 23, 2011
- CR-PLP-2011-01913, Inconsistency Between Micro ALARA Plan and RWP, April 18, 2011
- CR-PLP-2011-03273, Emergent Work Not discussed with ALARA Subcommittee, June 29, 2011
- EN-RP-105, Radiological Work Permits, Revision 9
- EN-RP-110, ALARA Program, Revision 7
- LO-PLPLO-2009-00047, LaSalle Station Collective Radiation Exposure, January 11, 2010 – January 13, 2010

- LO-WTPLP-2011-00084, CA-33, Develop SAC Agenda to Capture Information Discussed during the Outage, August 4, 2011
- LO-WTPLP-2011-00084, CA-34, Develop Method to Validate Dose Estimates for Outage with SRMP Data Points and Other Survey Data, August 4, 2011
- LO-WTPLP-2011-00084, CA-35, Review the Hot Spot Program Criteria, August 4, 2011
- LO-WTPLP-2011-00084, CA-36, Develop Training Module to Instruct ALARA Staff on Work In-progress Reviews, August 4, 2011
- LO-WTPLP-2011-00184, ALARA Suggestion Program Deficiencies, May 23, 2011
- Palisades Nuclear Power Plant Five Year Dose Reduction Plan, 2011-2015, Revision 0
- PLPLO-2009-00121, ALARA Program, March 1, 2010 – March 5, 2010
- Radiation Work Permit (RWP) and Associated ALARA Files, RWP 2010435, Refuel Project Shielding, various dates
- Radiation Work Permit (RWP) and Associated ALARA Files, RWP 2010468, Pressurizer Spray Controls Valves CV-1057 & CV-1059, various dates
- Radiation Work Permit (RWP) and Associated ALARA Files, RWP 2010471, ISI Alloy 600 FAC Exams in Containment, various dates
- Snapshot Assessment of 1R20 Contingency Plans
- WI-RSD-A-0116, Micro ALARA Planning, Revision 5

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

- 2010 Palisades Respiratory Protection Focused Self-Assessment, November 5 - 19, 2010
- CR-PLP-2010-04731, HEPA Unit by P-50C Does Not Have Current Inspection Tag, October 9, 2010
- CR-PLP-2010-04768, Reactor Cavity HEPA Found Exhausting Air Instead of Pulling Air, October 10, 2010
- CR-PLP-2011-01445, HEPA Units Utilized in the SFP Tent and Boric Acid Storage Tank Room Are Not Listed in HEPA Vacuum Issue Log, March 24, 2011
- EN-RP-502, Inspection and Maintenance of Respiratory Protection Equipment, Revision 7
- EN-RP-504, Breathing Air, Revision 3
- PLLP-FBT-SCOTTSCBA75, Scott Air-Pak 75 Self-Contained Breathing Apparatus, Revision 1
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001674, November 22, 2010
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001674, November 12, 2009
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001674, November 21 – 22, 2008
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001688, November 23, 2010
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001688, November 10, 2009
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001688, November 19, 2008
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001322, November 23, 2010
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001322, November 9, 2009
- Scott PosiChek3 Test Results, Scott Air-Pak 75 4500, Serial Number 115S0751001322, November 18, 2008
- Work Order 52276009 01, Analysis of Breathing Air Systems, January 17, 2011
- Work Order 52317965 01, Analysis of Breathing Air Systems, June 27, 2011

4OA1 Performance Indicator Verification

- EN-LI-114; Performance Indicator Process; Revision 4
- NRC Indicator Occupational Exposure Control Effectiveness (OR-1), July 2010 through June 2011.
- NRC Indicator Reactor Coolant System (RCS) Specific Activity (BI-1), July 2010 through June 2011.
- Palisades MSPI Basis Document, June 26, 2008
- RHR MSPI validation packages, third quarter 2010 through second quarter 2011
- Selected Operator logs, July 2010 through June 2011

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

- EP-01, Drill and Exercise Performance Indicators, August 23, 2011
- ONP-12, Acts of Nature, Revision 28
- ONP-23.1, Primary Coolant Leak, Revision 25
- SEP, Site Emergency Plan, Revision 21
- Shift Hourly trends 1220 through 1430, selected parameters, 23 August 2011

4OA5 Other Activities

- CR-PLP-0
- CR-PLP-05631, Failure to Update SAMGs, October 26, 2011
- CR-PLP-2007-01947, Condition Report, May 10, 2007
- EN-DC-115, Engineering Change Process, Revision 11
- EN-DC-115, Engineering Change Process, Revision 12
- FP-E-MOD-02, Engineering Change Process, Revision 2
- FP-E-MOD-04, Regulatory Commitments, Revision 2
- FP-E-MOD-05, Modification Plant Impact, Revision 2
- FP-E-MOD-11, Control of Design Interfaces, Revision 2
- LTR-PSA-07-08, Evaluation of Postulated Spurious Operation that Could Result in Opening of CV-1359
- M-213, P&ID – Service Water Screen Structure, Revision 90
- ONP-25.2, Off-Normal Procedure – Alternate Safe Shutdown, Revision 24
- SAMG, Severe Accident Management Guidelines, Revisions 2 and 3

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
AFW	Auxiliary Feedwater
ALARA	As-Low-As-Is-Reasonably-Achievable
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CV	Control Valve
DC	Direct Current
EAL	Emergency Action Level
IMC	Inspection Manual Chapter
IP	Inspection Procedure
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPO	Nuclear Plant Operator
NRC	U.S. Nuclear Regulatory Commission
PARS	Publicly Available Records System
PCS	Primary Coolant System
PI	Performance Indicator
RP	Radiation Protection
RPS	Reactor Protection System
RPT	Radiation Protection Technician
RWP	Radiation Work Permit
SAMG	Severe Accident Management Guidelines
SDP	Significance Determination Process
SSD	Safe Shutdown
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
UE	Unusual Event
URI	Unresolved Item
WO	Work Order

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Sincerely,

/RA/

John B. Giessner, Chief
Branch 4
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Letter to A. Vitale from J. Giessner dated November 8, 2011.

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