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

May 15, 2008

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

SUBJECT: PALISADES NUCLEAR PLANT NRC INTEGRATED INSPECTION REPORT 05000255/2008002

Dear Mr. Schwarz:

On March 31, 2008, the U. S. Nuclear Regulatory Commission (NRC) completed an inspection at your Palisades Nuclear Plant. The enclosed report documents the inspection findings which were discussed on April 23 with you and 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.

The report documents four NRC-identified findings and five self-revealed findings of very low safety significance (Green). Eight of these findings were determined to involve violations of NRC requirements. Additionally, two licensee identified violations which were determined to be of very low safety significance are listed in the report. However, because the violations were of very low safety significance and because the issues have been entered into your corrective action program, the NRC is treating these findings as Non-Cited Violations (NCVs) consistent with Section VI.A.1 of the Enforcement Policy.

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

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

Sincerely,

/RA/

Christine A. Lipa
Branch 4
Division of Reactor Projects

Docket No. 50-255; 72-007
License No. DPR-20

Enclosure: Inspection Report 05000255/2008002
w/Attachment: Supplemental Information

cc w/encl: Senior Vice President
Vice President Oversight
Senior Manager, Nuclear Safety & Licensing
Senior Vice President and COO
Assistant General Counsel
Manager, Licensing
W. DiProfio
W. Russell
G. Randolph
Supervisor, Covert Township
Office of the Governor
T. Strong, Chief, State Liaison Officer, State of Michigan
Michigan Department of Environmental Quality -
Waste and Hazardous Materials Division
Michigan Office of the Attorney General

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Letter to C. Schwarz from C. Lipa dated May 15, 2008

SUBJECT: PALISADES NUCLEAR PLANT NRC INTEGRATED INSPECTION REPORT 05000255/2008002

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket Nos.:	50-255; 72-007
License No.:	DPR-20
Report No.:	05000255/2008002
Licensee:	Entergy Nuclear Operations, Inc.
Facility:	Palisades Nuclear Plant
Location:	Covert, MI
Dates:	January 1 through March 31, 2008
Inspectors:	J. Ellegood, Senior Resident Inspector J. Giessner, Resident Inspector S. Bakhsh, Reactor Inspector, Decommissioning Branch J. Cassidy, Health Physicist M. Learn, Nuclear Safety Professional, Decommissioning Branch D. McNeil, Senior Operations Engineer S. Orth, Health Physics Team Leader R. Walton, Operations Engineer
Technical Specialist:	P. Lee, PhD., Health Physicist, Decommissioning Branch
Approved by:	C. Lipa, Chief Branch 4 Division of Reactor Projects

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000255/2008002, 01/01/2008 – 03/31/2008; Palisades Power Plant; Fire Protection, Maintenance Effectiveness, Outage Activities, Access Control to Radiologically Significant Areas, Radiological Environmental Monitoring Program and Radioactive Material Control Program, Follow up of Events and Notices of Enforcement Discretion

This report covers a three-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. This report includes nine Green findings, eight of which were non-cited violations (NCVs). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after U.S Nuclear Regulatory Commission (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-Revealing Findings

Cornerstone: Initiating Events

- Green. A self-revealed finding occurred on January 13 when the 'B' Main Feed Pump failed. The failure occurred due to improper maintenance on the lube oil pump associated with the Main Feed Pump that resulted in a loss of lube oil flow and trip of the Main Feed Pump. The failure was not a violation of NRC requirements. The licensee manually tripped the reactor in accordance with procedures and repaired the Main Feed Pump. The licensee entered the issue into the corrective action program as Condition Report (CR)-PLP-2008-0151 and repaired the pump.

  The inspectors concluded that this finding is more than minor in accordance with Inspection Manual Chapter 0609 because the finding is associated with the increase in the likelihood of an initiating event. Specifically, the improper pump assembly led to a partial loss of feed and subsequent plant trip. The inspectors determined the finding is of very low safety significance, Green, in accordance with the phase one screening checklist because the finding did not affect a mitigating system in addition to being a transient initiator. The finding does not represent a violation of NRC requirements; however, it does represent a failure to meet self imposed requirements to provide task instructions commensurate with the complexity of the work and qualifications of the workers. The finding includes a cross-cutting aspect in the area of Human Performance, resources due to an inadequate work package (H.2(c)). (Section 4OA3)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a Green NCV of License Condition 2.C.(3), Fire Protection, for failure to ensure a fire door between an emergency diesel generator room and a vital switchgear room was closed. This partially open door degraded the fire containment capability assumed in the fire hazards analysis. The fire door was closed and this issue was entered into the licensee's corrective action program as CR-PLP-2008-00075.

The finding is more than minor because it is associated with the protection against external factors (fires) attribute of the mitigating system cornerstone and affected the objective to maintain the reliability and capability of systems that respond to events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, Appendix F, Fire Protection SDP, the inspectors conducted a Phase I SDP screening. The inspectors determined the finding is of very low safety significance (Green), because the fire areas had fully functional, automatic water-based fire suppression which provided adequate coverage in both rooms and no transient combustible loads were present in either room. The finding includes a cross-cutting aspect in the area of human performance in that human error prevention techniques (H.4(a)), in this case adequate self checking, were not effective in ensuring this door was closed after use. (Section 1R05)

- Green. The inspectors identified a Green NCV of Title 10, Code of Federal Regulations (CFR) 50.65 for the failure to include a 'B' feed regulating valve deficiency to close during startup operations as a functional failure in the maintenance rule program. The inspectors noted that the failure should have placed the feedwater system into maintenance rule 10 CFR 50.65a(1) status in the fourth quarter of 2007. This caused a lapse in the determination of appropriate system monitoring and goal setting to maintain system reliability. This issue was entered into the licensee's corrective action program as CR-PLP-2008-00562 and the licensee placed the system in a(1) status.

  The finding is more than minor because, in accordance with Inspection Manual Chapter 0612, Appendix E, Examples of Minor Issues (example 7b) and Enforcement Manual Section 8.1.11, Maintenance Rule a(1) and a(2) violations are not minor because they involve structures, systems, and components (SSCs) that have demonstrated some degraded performance or condition. The finding is of very low safety significance because there was no design deficiency, the finding did not represent an actual loss of a safety function, nor does this involve a risk significant system for mitigating fire, flood, seismic, or severe weather events. This finding also has a cross-cutting aspect in the area of problem identification and resolution associated with the corrective action program (P.1(c)) because the licensee failed to thoroughly evaluate the cause and extent of condition of the failed feed regulating valve. (Section 1R12)

- Green. The inspectors identified a NCV of Technical Specification (TS) 5.4.1 for the failure to have adequate procedure guidance for the general operating procedures for mode transition to power operations. Specifically the general plant operating procedure for mode transition did not have adequate guidance to ensure the actions required by TS 3.0.4 were completed for failure of a radiation monitor required by TS prior to mode transition. Prior to the mode transition, the licensee completed the required action based on the inspectors' concerns and wrote a CR.

  The inspectors determined the failure to have adequate procedures for mode transition in accordance with TS is more than minor because, if left uncorrected, this and other mode transitions could have occurred with less than the required equipment operable or appropriate actions completed, which could become a more significant safety concern. The inspectors determined the finding is of very low safety significance (Green), because the actual mode transition occurred only after completion of the required actions based on the response to the inspectors' concerns. The finding includes a cross-cutting aspect in the area of human performance in that licensee did not adequately use conservative assumptions in decision-making to demonstrate the proposed action was safe (H.1(b)). (Section 1R20)

- Green. A self revealing NCV of TS 3.8.4 B and C was identified for failing to recognize that battery cell parameters were not within TS limits and for failing to take actions in accordance with TS for an inoperable battery. Specifically, cell 43 of the right train safety-related battery (ED02) was below technical specifications for individual cell voltage without recognition by the site staff. As a result, compensatory actions and a plant shutdown required by TSs were not completed as required. As an immediate action, the licensee completed the required actions required by TS including restoration of the battery to an operable status.

  The inspectors determined the failure to take required actions in accordance with TSs is more than minor because the finding impacts the equipment performance attribute of the Mitigating Systems cornerstone and adversely affects the objective to ensure availability, reliability and capability of the systems which respond to initiating events. The inspectors determined the finding is of very low safety significance (Green), because the finding did not cause a loss of safety function for the right train battery. The finding includes a cross-cutting aspect in the area of human performance in that human error prevention techniques (H.4(a)), in this case an adequate pre-job brief, were not effective in ensuring prompt notification of the shift manager. (Section 4OA3)

- Green. A self revealing NCV of TS 3.5.2 B and C was identified for the inability of an automatic valve in the Emergency Core Cooling System (ECCS), CV-3047, to reposition fully closed on an actuation signal. As a result, one train of ECCS was inoperable for longer than allowed by technical specifications. When the licensee identified that the valve would not fully close, the licensee took the actions required by TS and repaired the valve.

  The inspectors determined the failure to take required actions in accordance with TSs is more than minor because the finding impacts the equipment performance attribute of the Mitigating Systems cornerstone and adversely affects the objective to ensure availability, reliability and capability of systems which respond to initiating events. During the injection phase of an accident, more flow would bypass the core with the valve approximately 18 percent open, than if the valve had been fully closed. The inspectors determined the finding is of very low safety significance (Green) because the finding is not associated with a loss of safety function for the ECCS system. The finding includes a cross-cutting aspect in the area of human performance in that the licensee did not adequately coordinate work activities to address the impact of actions needed to ensure the valve was closed when the valve was declared inoperable (H.3(b)). (Section 4OA3)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a finding of very low safety significance and associated NCV of 10 CFR 20.1703(c) for the failure to implement written procedures to ensure batteries for powered air purifying respirators (PAPRs) are adequately charged before use. As of January 16, 2008, the licensee failed to maintain procedures that provided adequate instructions concerning the charging of PAPR batteries, which resulted in two failures of a PAPR unit to properly function and in the intake of radioactive material on September 9, 2007. As corrective actions, the licensee revised procedures and replaced the battery chargers with a model that indicates battery charge condition. The licensee entered the issue into the corrective action program as CR-PLP-2007-04149 and CR-PLP-2008-00229.

The finding is more than minor because it impacted the equipment and instrumentation attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation, in that not providing adequate procedures for control of PAPR battery charging resulted in an unplanned exposure to radioactive material. The finding was determined to be of very low safety significance because it was not an As Low As Reasonably Achievable (ALARA) planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised. The inspectors did not identify a cross-cutting aspect associated with this finding. (Section 2OS1.2)

- Green. A self-revealed finding of very low safety significance and associated NCV of 10 CFR 20.1701 was identified for the failure to use, to the extent practical, process or other engineering controls to control the concentration of radioactive material in air. On September 12, 2007, the licensee failed to implement effective engineering controls in the reactor containment to reduce the levels of radioactive iodine gases. The failure resulted in elevated levels of airborne radioactivity and the intakes of radioactive material by the licensee's staff. As corrective actions, the licensee conducted a root cause evaluation and has entered the problem in the corrective action program as CR-PLP-2007-04002.

  The finding is more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation, in that not implementing adequate engineering controls resulted in unplanned exposures to radioactive material. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised. The engineering controls comprised of a charcoal filtration ventilation system that were planned to be used to control the concentration of radioactive material in air were either depleted soon after placed in service or installed backwards. Consequently, the cause of this deficiency had a cross-cutting aspect (H.3(a)) in the area of Human Performance related to work control. Specifically, the licensee failed to plan and coordinate work activities with planned contingencies and compensatory actions. (Section 2OS1.2)

Cornerstone: Public Radiation Safety

- Green. A self-revealed finding of very low safety significance and associated NCV of 10 CFR 20.1501 was identified for failure to perform an adequate radiological survey to assure compliance with 10 CFR 20.1802, which requires that the licensee control and maintain constant surveillance of licensed material that is in a controlled area or unrestricted areas and that is not in storage. On January 17, 2008, the NRC notified the licensee that radioactive material was identified by another NRC licensed facility when workers arrived following Palisades refueling outage 1R19. That licensee identified six pairs of footwear and other personal items with radioactive contamination levels between 6,000 and 20,000 disintegrations per minute, which had been improperly released from the Palisades site. As immediate corrective actions, the affected materials were confiscated by the other site. Additionally, the licensee identified two earlier occurrences of inappropriate surveys that were performed early in the refueling outage that resulted in the inadvertent release of radioactive material. As corrective actions, the licensee planned to implement new procedure documents, and the issue was entered into the licensee's corrective action program as Condition Reports CR-PLP-2007-04338 and CR-PLP-2008-01180.

The finding is more than minor because it impacted the program and process attribute of the Public Radiation Safety Cornerstone and it adversely affected the cornerstone objective of ensuring adequate protection of public health and safety from exposure to radioactive material released into the public domain, in that inadequate surveys resulted in the failure to control radioactive material. The finding was determined to be of very low safety significance because it was a radioactive material control finding, it was not a transportation finding, and it did not result in public dose greater than 0.005 rem. The finding was caused by the decision to allow manual release surveys of a large number of workers that alarmed the personal contamination monitor, which overwhelmed the ability of the radiation protection staff to conduct effective monitoring of personnel. Consequently, the cause of this deficiency had a cross-cutting aspect (H.1(a)) in the area of Human Performance related to decision making. Specifically, the licensee failed to make risk-significant decisions using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety is maintained. (Section 2PS3.1)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period at or near 100 percent reactor power. On January 13, the plant was manually tripped by control room licensed operators due to the loss of the B main feed pump. On January 14, the licensee restarted the plant and proceeded to 80 percent power. On January 18, the licensee completed repairs on the B main feed pump and raised power to 100 percent. The plant remained at or near 100 percent for the remainder of the quarter.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Evaluation of Readiness for Impending Adverse Weather Conditions

a. Inspection Scope

The inspectors completed one adverse weather site sample. The inspectors reviewed the site's procedure and readiness for high winds and extremely cold weather on February 10-11, 2008. The inspectors reviewed the licensee's procedures to determine if actions specified could be completed with expected plant staffing and to determine if areas identified would be accessible. In addition, the inspectors compared proceduralized actions with the Updated Final Safety Analysis Report to determine if vulnerabilities existed. The inspectors reviewed the licensee's response to fouling in the service water strainers; and verified that operator actions defined in the licensee's adverse weather procedure maintained readiness of essential systems.

These activities constituted one adverse weather sample as defined by Inspection Procedure 71111.01.

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Left train Auxiliary Feed Water (AFW) with pump P-8C out for scheduled work
- Right train AFW with pump P-8A out for scheduled work
- Left train High Pressure Safety Injection (HPSI) with right train out for scheduled work

The inspectors selected these systems based on their risk significance relative to the reactor safety cornerstones at the time they were inspected. The inspectors focused on 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, Technical Specification (TS) requirements, Administrative TS, outstanding work orders, condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components and support equipment were aligned correctly and operable. The inspectors examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors also verified that the licensee had properly identified and resolved equipment alignment problems that could cause initiating events or impact the capability of mitigating systems or barriers and entered them into the corrective action program with the appropriate significance characterization. Documents reviewed are listed in the attachment.

These activities constituted three partial system walkdown samples as defined by Inspection Procedure 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on evaluating the material condition of active and passive fire protection features as well as control of transient combustible material. The inspectors walked down the following risk-significant plant areas:

- 1-2 Emergency Diesel Generator (EDG) fuel oil day tank room (Fire Area 8)
- ED01 battery room (Fire Area 12)
- ED02 battery room (Fire Area 11)
- C switch gear room (Fire Area 4)
- 590 foot turbine building
- Main transformer and startup transformer deluge system

The inspectors reviewed areas to assess if the licensee had implemented their fire protection program to control combustibles and ignition sources within the plant. Where installed, the inspectors verified that fire detection and suppression equipment was in good material condition. The inspectors verified that passive fire protection features were in good material condition. In cases where the licensee implemented a compensatory measure, the inspectors verified the compensatory measure was in accordance with the licensee's administrative procedure and was in effect. 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. The inspectors verified that fire hoses and extinguishers were 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 corrective action program.

These activities constituted six quarterly fire protection inspection samples as defined by Inspection Procedure 71111.05-05.

b. Findings

Introduction: The inspectors identified a Green NCV of License Condition 2.C.(3), Fire Protection, for failure to ensure a fire door between an emergency diesel generator room and a vital switchgear room was closed. This partially open door degraded the fire containment capability assumed in the fire hazards analysis.

Description: On January 8, 2008 while conducting a tour, the inspectors noted door 71, the fire door between the C bus safety-related switchgear room and the 1-1 EDG room, open about two inches. The fire door is a three-hour door which separates Fire Area 4 from Fire Area 5. Although there is an auto-closure mechanism on the door, when the ventilation system cycles on, the door will not close without assistance. This known condition is stated on a sign which is affixed to the door which says: "Attention Varying Air Pressures Affect Door Closing Please Manually Close Door Completely." In this case, the door was found partially open, and the ventilation fan was running in the 1-1 EDG room, resulting in the door being held partially open. The inspector saw no one in the immediate vicinity and closed the door. The inspectors looked in the adjacent vital areas and found no one. The inspectors informed the operations shift of the issue and the shift initiated CR-PLP-2008-00075.

The investigation determined the last known entry was 12 minutes earlier by security personnel conducting fire tours for unrelated issues. The inspectors concluded the fire door was not closed and should have been closed in accordance with the licensee's fire hazards analysis to provide a three hour fire barrier between a 2400v vital bus and an emergency diesel generator.

Analysis: The failure of an automatic fire door to close and the failure to close the door is a performance deficiency that warrants a significance determination. The inspectors reviewed the minor examples in Inspection Manual Chapter 0612, Appendix E, "Examples of Minor Issues," and none were found which related to this issue. The finding is more than minor because it is associated with the protection against external factors (fires) attribute of the mitigating system cornerstone and affects the objective to maintain the reliability and capability of systems that respond to events to prevent undesirable consequences. In accordance with Inspection Manual Chapter 0609, Appendix F, "Fire Protection SDP", the inspectors conducted a Phase I SDP screening. The inspectors determined this finding was in the fire confinement category and the barrier was moderately degraded because the door was not latched and was partially open. The inspectors determined the finding was of very low safety significance

(Green), because both fire areas had fully functional, automatic water-based fire suppression which provided adequate coverage in both rooms. No transient combustible loads were present in either room. The finding includes a cross-cutting aspect in the area of human performance in that human error prevention techniques (H.4(a)), in this case adequate self checking, were not effective in ensuring this door was closed after use.

Enforcement: Palisades License Condition 2.C.(3), Fire Protection, states, in part, that the licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report and approved in various Safety Evaluation Reports. Updated Final Safety Analysis Report chapter 9.6, "Fire Protection", states, in part, that building structures have been designed and arranged to prevent the spread of fire. The Updated Final Safety Analysis Report references the complete description of fire areas and barriers as being contained in the Fire Hazards Analysis Report. The Fire Hazards Analysis Report, Revision 7, requires fire barrier protection between Fire Areas 4 and 5 with three-hour fire walls and three-hour doors. Contrary to this, on January 8, 2008, licensee personnel failed to assure that openings in the fire barrier walls were protected with doors with a rating equivalent to that of the barriers. Specifically, door 71 was partially open and unlatched which made the fire door inoperable and invalidated the three hour fire rating of the fire barrier. The corrective actions to restore compliance included immediately ensuring the door was properly closed and latched. Because the finding is of very low safety significance and has been entered into the licensee's corrective action process as CR-PLP-2008-00075, this violation is being treated as an NCV consistent with Section VI.A of the Enforcement Policy: NCV 05000255/2008002-01, "Failure to Ensure Fire Door Was Closed."

1R11 Licensed Operator Requalification Program

.1 Facility Operating History (71111.11B)

a. Inspection Scope

The inspectors reviewed the plant's operating history from April 2006 through January 2008 to identify operating experience that was expected to be addressed by the Licensed Operator Requalification Training (LORT) program. The inspector verified that the identified operating experience had been addressed by the facility licensee in accordance with the station's approved Systems Approach to Training (SAT) program to satisfy the requirements of 10 CFR 55.59(c). The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.2 Licensee Requalification Examinations

a. Inspection Scope

The inspectors performed an inspection of the licensee's LORT test/examination program for compliance with the station's SAT program which would satisfy the requirements of 10 CFR 55.59(c)(4). The reviewed operating examination material consisted of six operating tests, each containing two-four dynamic simulator scenarios and five-six job performance measures (JPMs). The written examinations reviewed consisted of five written examinations, each consisting of only a Part B, Administrative Controls/Procedure Limits section. Each examination contained approximately 30 questions. The inspectors reviewed the annual requalification operating test and biennial written examination material to evaluate general quality, construction, and difficulty level. The inspectors assessed the level of examination material duplication from week-to-week during the current year operating test. The examiners assessed the amount of written examination material duplication from week-to-week for the written examination administered in 2007. The inspectors reviewed the methodology for developing the examinations, including the LORT program two-year sample plan, probabilistic risk assessment insights, previously identified operator performance deficiencies, and plant modifications. The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.3 Licensee Administration of Requalification Examinations

a. Inspection Scope

The inspectors observed the administration of a requalification operating test on February 7-8, 2008, to assess the licensee's effectiveness in conducting the test to ensure compliance with 10 CRF 55.59(c)(4). The inspectors evaluated the performance of one crew in parallel with the facility evaluators during two dynamic simulator scenarios and evaluated various licensed crew members concurrently with facility evaluators during the administration of several JPMs. The inspectors assessed the facility evaluators' ability to determine adequate crew and individual performance using objective, measurable standards. The inspectors observed the training staff personnel administer the operating test, including conducting pre-examination briefings, evaluations of operator performance, and individual and crew evaluations upon completion of the operating test. The inspectors evaluated the ability of the simulator to support the examinations. A specific evaluation of simulator performance was conducted and documented under Section 1R11.9, "Conformance with Simulator Requirements Specified in 10 CFR 55.46," of this report. The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.4 Examination Security

a. Inspection Scope

The inspectors observed and reviewed the licensee's overall licensed operator requalification examination security program related to examination physical security (e.g., access restrictions and simulator considerations) and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors also reviewed the facility licensee's examination security procedure, any corrective actions related to past or present examination security problems at the facility, and the implementation of security and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the examination process. The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.5 Licensee Training Feedback System

a. Inspection Scope

The inspectors assessed the methods and effectiveness of the licensee's processes for revising and maintaining its LORT Program up to date, including the use of feedback from plant events and industry experience information. The inspectors reviewed the licensee's quality assurance oversight activities, including licensee training department self-assessment reports. The inspectors evaluated the licensee's ability to assess the effectiveness of its LORT program and their ability to implement appropriate corrective actions. This evaluation was performed to verify compliance with 10 CFR 55.59(c) and the licensee's SAT program. The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.6 Licensee Remedial Training program

a. Inspection Scope

The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the previous biennial requalification examinations and the training from the current examination cycle to ensure that they addressed weaknesses in licensed operator or crew performance identified during training and plant operations. The inspectors reviewed remedial training procedures and individual remedial training plans. This evaluation was performed in accordance with 10 CFR 55.59(c) and with respect to the licensee's SAT program. The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.7 Conformance With Operator License Conditions

a. Inspection Scope

The inspectors reviewed the facility's and individual licensed operator's conformance with the requirements of 10 CFR Part 55. The inspectors reviewed the facility licensee's program for maintaining active operator licenses and to assess compliance with 10 CFR 55.53(e) and (f). The inspectors reviewed the procedural guidance and the process for tracking on-shift hours for licensed operators and which control room positions were granted watch-standing credit for maintaining active operator licenses. The inspectors reviewed the facility licensee's LORT program to assess compliance with the requalification program requirements as described by 10 CFR 55.59(c). Additionally, medical records for 12 licensed operators were reviewed for compliance with 10 CFR 55.53(I). The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.8 Conformance With Simulator Requirements Specified in 10 CFR 55.46

a. Inspection Scope

The inspectors assessed the adequacy of the licensee's simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements as prescribed in 10 CFR 55.46, "Simulation Facilities." The inspectors also reviewed a sample of simulator performance test records (i.e., transient tests, malfunction tests, steady state tests, and core performance tests), simulator discrepancies, and the process for ensuring continued assurance of simulator fidelity in accordance with 10 CFR 55.46. The inspectors reviewed and evaluated the discrepancy process to ensure that simulator fidelity was maintained. Open simulator discrepancies were reviewed for importance relative to the impact on 10 CFR 55.45 and 55.59 operator actions as well as on nuclear and thermal hydraulic operating characteristics. The inspectors conducted interviews with members of the licensee's simulator staff about the configuration control process and completed the Inspection Procedure 71111.11, Appendix C, checklist to evaluate whether or not the licensee's plant-referenced simulator was operating adequately as required by 10 CFR 55.46(c) and (d). The documents reviewed during this inspection are listed in the attachment.

b. Findings

No findings of significance were identified.

.9 Annual Operating Test Results

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the individual JPM operating tests, and the simulator operating tests (required to be given per 10 CFR 55.59(a)(2)) administered by the licensee from January 2008 through February 2008 as part of the licensee's operator licensing requalification cycle. These results were compared to the thresholds established in Inspection Manual Chapter 0609, Appendix I, "Licensed Operator Requalification Human Performance SDP." The evaluations were also performed to determine if the licensee effectively implemented operator requalification guidelines established in NUREG 1021, Operator Licensing Examination Standards for Power Reactors, and Inspection Procedure 71111.11, Licensed Operator Requalification Program. The documents reviewed during this inspection are listed in the attachment.

This review represented one biennial licensed operator requalification inspection sample.

b. Findings

No findings of significance were identified.

.10 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

On January 17 the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations 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
- the ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constitutes one quarterly licensed operator requalification program sample as defined in Inspection Procedure 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

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

- Feed Water System
- Primary Coolant System

The inspectors reviewed events where ineffective equipment maintenance has resulted in 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 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 corrective action program with the appropriate significance characterization. Documents reviewed are listed in the Attachment.

This inspection constitutes two quarterly maintenance effectiveness samples as defined in Inspection Procedure 71111.12-05.

b. Findings

Introduction: The inspectors identified a Green NCV of 10 CFR 50.65(a)(1) for the failure to include a 'B' feed regulating valve deficiency to close during startup operations as a functional failure in the maintenance rule program. The inspectors noted that the failure would have placed the feedwater system into maintenance rule 10 CFR 50.65(a)(1) status in the fourth quarter of 2007. The failure to properly categorize the failure of the valve to close resulted in a delay in establishing appropriate system monitoring and goal setting to maintain system reliability.

Description: The inspectors reviewed the apparent cause for a plant transient that occurred on October 20, 2007. While the plant was in mode 2 at about 3 percent power following a refueling outage, the operations' staff attempted to transfer from auxiliary feedwater to main feedwater. When the B feed regulating valve, CV-0703, was un-isolated, primary coolant temperature dropped and steam generator level began to rise.

Although CV-0703 was believed to be closed, it was partially open. Temperature dropped to within .3 degrees Fahrenheit of the minimum temperature for critical operations required by TS and Steam Generator level rose to 86 percent. The operations staff backed out of the procedure, isolated the valve, and took action to repair the valve.

The licensee determined that during the outage, maintenance testing on the valve positioner caused the bias spring to shift and offset the zero for the valve positioner. As a result, the valve remained partially open even though the control signal demanded a full close position. Even though the post maintenance test did not detect the condition the cause evaluation did not evaluate why the post maintenance test failed to detect this deficiency. In addition, the apparent cause determined that the condition did not affect any maintenance rule functions. The inspectors reviewed the maintenance rule scoping document and found the valve's closing function is listed in the scoping document. The inspectors provided this information to engineering and engineering wrote CR 2008-00562. On February 5, 2008, the inspectors reviewed the system health report of record dated July 11, 2007. The report identified there was 1 maintenance preventable functional failure in the previous 24 months; and established a performance criterion of <2 maintenance preventable functional failure in a 24 month period. One additional maintenance preventable functional failure would place the system in a(1) status. On February 27, 2008, the expert panel met and determined the failure of B feed regulation valve CV-0703 was a maintenance preventable functional failure. In addition, the panel reviewed this maintenance preventable functional failure and subsequent items and placed the system in a(1) status.

<u>Analysis:</u> The inspectors concluded that the failure to categorize the B feed regulating valve failure to close as a maintenance preventable functional failure was a performance deficiency and warranted an assessment in the SDP. The inspectors determined that once the licensee included the valve's failure to close as a maintenance preventable functional failure, the system should have been placed in a(1). Because of the failure to properly categorize the failure, the licensee delayed placement of the system into a(1) for several months. The issue is more than minor because, in accordance with MC 0612, Appendix E, Examples of Minor Issues (example 7b) and Enforcement Manual Section 8.1.11, Maintenance Rule a(1) and a(2) violations are not minor because they involve SSCs that have demonstrated some degraded performance or condition. The finding is of very low safety significance because there was no design deficiency, the finding did not represent an actual loss of a safety function, nor does this involve a risk significant system for mitigating fire, flood, seismic, or severe weather events. This finding also had cross-cutting aspects in the area of problem identification and resolution associated with the corrective action program (P.1(c)) because the licensee failed to thoroughly evaluate the cause and extent of condition of the failed feed regulating valve.

<u>Enforcement:</u> 10 CFR 50.65, "Maintenance Rule", paragraph a(1) states, in part, that the performance or condition of systems shall be monitored against established goals to provide reasonable assurance that the systems are capable of performing their intended functions. Paragraph a(2) of 10 CFR 50.65 requires, in part, that monitoring as specified in paragraph a(1) is not required where it has been demonstrated that the performance or condition of a system is being effectively controlled through the performance of appropriate preventive maintenance such that the system remains capable of performing its intended function. Contrary to the above, although the licensee had sufficient

information on November 11, 2007, (the date the cause evaluation indicated the failure was not a maintenance preventable functional failure) to classify the failure to close as a maintenance preventable functional failure, the licensee failed to properly evaluate the system under the maintenance rule process. This resulted in a delay in monitoring performance of the main feedwater system to provide assurance that the maintenance on the system was effective in maintaining the system capable of performing its intended function. Specifically, the inspectors determined that the performance of the main feedwater system was such that it was necessary to monitor system performance against established goals under a(1) when an additional functional failure occurred for B feed regulating valve CV-0703. The licensee failed to place the system in a(1) and therefore failed to establish goals and/or monitor the performance of the system against such goals. The failure to establish goals and monitor feedwater system under a(1), is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000255/2008002-02: Failure to Monitor the Feedwater System Under 10 CFR 50.65a(1) . This issue is in the licensee's corrective action program as CR-PLP-2008-00562. The licensee placed the system in a(1) status.

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

.1 Maintenance Risk Assessments and Emergent Work Control

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:

- Risk assessment with ED15 battery charger out of service for planned test
- Risk assessment with 1-1 EDG out of service for planned test and extended for emergent work
- Risk assessment for a scheduled electrical front bus outage
- Risk assessment with steam line for CV 0554 out of service for maintenance
- Risk Assessment for a planned AFW pump outage

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 activities constitute five samples as defined by Inspection Procedure 71111.13-05.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- ED02 (battery #2) cell 23 performance degrading
- Fuel handling ventilation and chiller performance for fuel movement
- 1-1 EDG common cause failure evaluation for 1-2 EDG material issue
- Relief Valve RV0719 fasteners missing and loose
- ED02 (battery #2) cell 37 performance degrading
- Low pressure safety injection isolation valve CV3006 not in configuration for remote operations
- Moisture Separator Reheater Drain Tank piping support fatigue failure

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 Updated Final Safety Analysis Report 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 also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the attachment.

This inspection constitutes seven samples as defined in Inspection Procedure 71111.15.-05

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

.1 Post Maintenance Testing

a. Inspection Scope

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

- ‘A’ service water pump following repack
- CV-3047, ‘C’ safety injection fill valve maintenance
- ‘B’ main feed pump maintenance
- Turbine driven AFW pump maintenance
- 1-2 EDG maintenance on February 21

These activities were selected based upon the SSC’s ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (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 TS, the Updated Final Safety Analysis Report, 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 corrective action program and that the problems were being corrected commensurate with their importance to safety. For the EDG sample, the inspectors used Review of Operating Experience Smart Sample: “OpESS FY2008-01, A Negative Trend and Recurring Events Involving Emergency Diesel Generators” for guidance in conducting the inspection. Documents reviewed are listed in the attachment.

This inspection constitutes five samples as defined in Inspection Procedure 71111.19.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Forced Outage Activities

a. Inspection Scope

The inspectors evaluated outage activities for a forced outage that began on January 13, 2008, and continued through to January 14, 2008. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed recovery actions following a manual reactor trip due to loss of the B main feed pump. The inspectors reviewed outage equipment configuration and risk management, electrical lineups, control and monitoring of decay heat removal, control of containment activities, and startup activities. The inspectors reviewed identification and resolution of problems identified during the outage.

This inspection constitutes one other outage sample as defined in Inspection Procedure 71111.20-05.

b. Findings

Introduction: The inspectors identified a Green NCV of TS 5.4.1 for the failure to have adequate procedure guidance for the general operating procedures for mode transition to power operations. Specifically, the general plant operating procedure for mode transition did not have adequate guidance to ensure the actions required by TS 3.0.4 were completed for a failure of a radiation monitor required by TS.

Description: On January 14, 2008 with the plant in mode 3, during startup inspection activities in the morning, the inspectors noted radiation monitor RIA-1805, a safety-related monitor, was listed on the Limiting Condition for Operation (LCO) board as being inoperable. RIA-1805 is one of four containment radiation monitors required by TS 3.3.3 (Function # 6 of table 3.3.3-1, applicable in modes 1-4). The monitors are part of engineered safety features and have a two out of four coincidence logic to actuate to isolate the containment based on high radiation. The inspectors questioned the operations team if the monitor was to be restored to operable prior to start-up (transition from mode 3 to mode 2) or placed in trip since TS 3.0.4, in general, required systems to be operable prior to an upward mode transition unless the actions entered allowed for unlimited period of time. The start-up was scheduled to occur within the hour. The operations shift indicated the issue had been reviewed by their on-site review committee, the Plant Oversight Review Committee and signed off as acceptable in General Operating Procedure (GOP), GOP-3, "Mode 3> 525F to Mode 2," step 1.14. The rationale was that the channel could be placed in trip and once the channel was placed in trip, the plant could be operated for an unlimited period of time. The site assumed they had seven days to place the unit in trip (the required completion time) and that it did not have to be completed prior to the mode ascension. The inspectors noted that since the action had not been taken (tripping the channel would change the coincidence logic from two out of four channels to one out of three channels to actuate the engineered safety features), the plant was still in a shutdown action statement. The required action has seven days to be completed, but if it is not completed or the time is not met, the plant must be shut down (action E of TS 3.3.3). Because of the inspectors' concerns, operations decided to complete the repair of the radiation channel. At 0532 the licensee declared RIA-1805 operable.

The reactor startup was delayed for reasons not related to RIA-1805; however, RIA-1805 failed again at 1303. After the other startup delays were resolved, with another operations shift in the control room, the reactor startup procedure was about to be started. The inspectors asked if they were planning to place the channel in trip or repair the channel prior to startup. The assistant operations manager indicated that the issue was previously reviewed by the Plant Oversight Review Committee and that there was no actual requirement to take the action which allows operations for an unlimited period of time prior to using TS 3.0.4a provision for mode transition. The inspectors discussed the issue with the shift manager as well. After discussing with plant management, the shift indicated they would place the channel in trip and then proceed with the startup. The shift determined the correct methodology for tripping the channel; tripped the channel at time 1423; and then proceeded with the start-up at 1426. The licensee wrote CR PLP-2008-00180 to address the issue.

The inspectors reviewed TS 3.0.4 and the basis for TS 3.0.4a and concluded that since the objective of TS 3.0.4 was to assure that adequate safety was maintained during mode ascension, the required actions must be completed prior to mode transition. While it would be optimum to have all equipment operable, TS 3.0.4a allows mode ascension if the actions to be entered allowed unlimited period of time. The basis says: "Compliance with the required Actions that permit continued operations of the plant for unlimited period of time in a mode or other specified condition provides an acceptable level of safety for continued operation." Since the action to change the coincidence from the engineered safety features actuation from two out of three (since one is failed) to one out of three, is the item which provides the acceptable level of safety, the inspectors concluded until the licensee completed the required action, TS 3.0.4a was not satisfied. The inspectors concluded the licensee's assessment was not accurate.

The inspectors requested the assistance of staff in the Region III and Nuclear Reactor Regulation Offices for the TS interpretation for TS 3.0.4 a. The single item to be addressed: Do the associated actions which permit continued operation for an unlimited period of time (in this case placing the bistable in the trip condition for radiation monitor, RI 1805, pursuant to TS 3.3.3 Action A) need to be completed before the mode transition from mode 3 to mode 2 occurs; or can it be done anytime in the seven day completion time? The group evaluated the TS and concurred in Task Interface Agreement (TIA) 2008-002, dated May 9, 2008, which validated for mode ascension the actions that allow operating for an unlimited period of time (i.e. placing the instrument in the tripped condition) must be completed prior to the mode ascension. Otherwise the plant remains in a shutdown LCO and the TS 3.0.4a can not be applied. The team concluded that the licensee was not properly applying TS 3.0.4.

The inspectors concluded that the licensee's assessment, including their sign-off in GOP-3, step 1.14 was not appropriate; and that a mode transition would have been conducted with RIA-1805 inoperable if the inspectors had not intervened. The inspector's review of the procedure determined the guidance in the GOP was not adequate to ensure the action, which subsequently allows unlimited operating time, was completed prior to mode transition.

Analysis: The inspectors determined the failure to have adequate procedures for mode transition to ensure compliance with TSs required a significance determination in accordance with Inspection Manual Chapter 0609. The minor examples of Inspection Manual Chapter 0612 Appendix E were reviewed. Example k was pertinent and provided an example of a minor item where there were not programmatic concerns which could lead to worse errors if uncorrected. Since validation of compliance with TS 3.0.4 is not adequately captured and multiple groups reviewed the issue; the inspectors concluded this issue was programmatic. Therefore, the finding is more than minor because, if left uncorrected, the finding would become a more significant safety concern in that the licensee would have transitioned modes in a manner prohibited by TS. The finding was considered to have very low safety significance (Green), because the correct actions were completed prior to mode transition based on the response to the inspectors' concerns. The finding included a cross-cutting aspect in the area of human performance in that the licensee did not adequately use conservative assumptions in decision-making to demonstrate the proposed action was safe (H.1(b)). Specifically, taking actions to restore systems to an operable status prior to mode transition is critical to conservative decision-making.

Enforcement: Technical Specification 5.4.1 requires, in part, that written procedures shall be established, implemented and maintained covering applicable procedures recommended in Appendix A of Regulatory Guide 1.33. Appendix A item 2a is an applicable written procedure for Hot Standby to Minimum Load (nuclear start-up). Procedure GOP-3, "General Operating Procedure Mode 3 > 525 to Mode 2," Revision 25 is the site's written procedure to conduct this evolution. Contrary to the above on January 14, 2008, the site's guidance for mode transition for review of TS 3.0.4 (specifically step 1.14) was not adequately maintained in that it did not provide adequate guidance to assess what actions need to be completed to ensure TS 3.0.4a could be applied. Because this finding was of very low safety significance, the finding was entered into the licensee's corrective action program as CR-PLP-2008-00180; this finding is being dispositioned as an NCV (NCV 0500255/2008002-03, Inadequate General Operating Procedure) consistent with Section VI.A of the NRC Enforcement Policy.

1R22 Surveillance Testing (71111.22)

.1 Routine Surveillance Testing

a. Inspection Scope

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

- 1-1 EDG surveillance testing
- DWO-1 plant calorimetric heat balance
- QI-9, Reactor Protective Trip Units
- Daily crane check prior to heavy load movement
- Alert and Notification System siren testing

The inspectors observed in-plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were 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 were in accordance with TSs, the Updated Final Safety Analysis Report, 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, 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 in the corrective action program. Documents reviewed are listed in the attachment.

This inspection constitutes five routine surveillance testing samples as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

.2 Inservice Testing

a. Inspection Scope

The inspectors reviewed the test results for the following activity 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:

- Inservice testing of 7C Service Water pump

The inspectors observed in plant activities and reviewed procedures and associated records to determine whether: preconditioning occurred; effects of the testing were adequately addressed by control room personnel or engineers prior to the commencement of the testing; acceptance criteria were clearly stated, demonstrated operational readiness, and were 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 Updated Safety Analysis Report, 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 corrective action program. Documents reviewed are listed in the attachment.

This inspection constitutes one inservice inspection sample as defined in Inspection Procedure 71111.22.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness [EP]

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on March 19, 2008, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator control room, Technical Support Center and Emergency Operating Facility to verify that event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to verify whether the licensee staff was properly identifying weaknesses and entering them into the corrective action program. As part of the inspection, the inspectors reviewed the drill package listed at the end of this report.

This inspection constitutes one drill sample as defined in Inspection Procedure 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety [OS]

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Review of Licensee Performance Indicators for the Occupational Exposure Cornerstone

a. Inspection Scope

The inspectors reviewed the licensee's occupational exposure control cornerstone Performance Indicators to determine whether the conditions resulting in any Performance Indicator occurrences had been evaluated and whether identified problems had been entered into the corrective action program for resolution.

This inspection constituted one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors reviewed the following radiation work permits (RWPs) for airborne radioactivity areas to verify barrier integrity and engineering controls performance (e.g., high-efficiency particulate air ventilation system operation) and to determine if there was a potential for individual worker internal exposures of >50 millirem committed effective dose equivalent:

- RWP 754; Refuel Project – Reactor Vessel Disassembly;
- RWP 823; Valve Repair Activities in Containment; and
- RWP 756; Refuel Project – Reactor Cavity Decontamination.

Work areas having a history of, or the potential for, airborne transuranics were evaluated to verify that the licensee had considered the potential for transuranic isotopes and provided appropriate worker protection.

This inspection constituted one sample as defined by Inspection Procedure 71121.01-5.

The adequacy of the licensee's internal dose assessment process for internal exposures due to the elevated containment airborne radioactivity during the fall 2007 refueling outage was assessed.

This inspection constituted one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

(1) Failure of Powered Air Purifying Respirator

Introduction: A Green NRC-identified finding of very low safety significance and associated NCV of 10 CFR 20.1703 was identified for the failure to maintain adequate written procedures regarding the storage, issuance, and maintenance of respiratory protection equipment.

Description: The reactor head O-ring was removed during the (1R19) refueling outage. This work was planned and controlled under RWP 754, Refuel Project – Reactor Vessel Disassembly. The RWP required the use of respiratory protection, specifically, a PAPR, for this evolution. On September 9, 2007, the reactor head O-ring was removed, as planned. However, during the job evolution the battery that supplied power to the PAPR failed while the respirator was being worn. The user immediately notified the radiation protection technician who replaced the battery, then the user continued to work. The second battery failed about one hour after it was placed in service. The second failure caused the worker to immediately exit the work area. The radiation protection technician observed that the worker exhibited signs of distress and took immediate actions to remove the PAPR quickly by tearing it down and away from the worker's head. The unordinary method of removal was required because of worker distress but contributed to the intake of radioactive material by the worker. The licensee performed an assessment of the worker's internal dose and verified the dose was well below regulatory limits.

The licensee performed an apparent cause evaluation and determined that the two failures of the PAPR were caused by incomplete charging of the batteries prior to being placed in service. The manufacturer of the battery charger provided instructions for battery maintenance, indicating that the battery should be charged for two times the length of the previous use. However, the licensee had not included this guidance in its procedures, training, or practice. Specifically, the licensee had not established a method to identify the length of time a battery was used or the length of time that the battery was charged. Additionally, the charger used by the licensee did not provide any indication whether the battery was fully charged.

The inspectors reviewed the corrective actions taken to prevent batteries from being issued before being completely charged. Specifically, the licensee's apparent cause evaluation recommended that the licensee purchase new chargers (dual rate chargers) and replace the older chargers used during the outage with the dual rate design. The dual rate chargers provided a light emitting diode to indicate that the battery is fully charged and ready for use. During the inspection, the inspectors observed most of the batteries were still being charged with the old style chargers after the corrective action was to have been completed. The inspectors informed the respiratory protection program owner of the corrective action and its scheduled completion date. The respiratory protection program owner removed all of old style chargers after validating this observation. Additionally, the licensee planned to revise respiratory protection procedures and training to prevent recurrence.

Analysis: The inspectors determined that this finding was a performance deficiency because licensees are required to adhere to the regulations contained in Subpart H of 10 CFR Part 20, which requires licensees to implement and maintain applicable respiratory protection procedures. The inspectors also determined that the performance deficiency was reasonably within the licensee's ability to foresee and correct. In accordance with NRC Inspection Manual Chapter 0612, the inspectors determined that the finding was more than minor because it impacted the equipment and instrumentation attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of ensuring adequate protection of worker health and safety from exposure to radiation, in that not providing adequate procedures for control of PAPR battery charging resulted in an unplanned exposure to radioactive material. The finding was assessed using the Occupational Radiation Safety Significance Determination Process and was determined to be of very low safety significance (Green) because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised.

The inspectors did not identify a cross-cutting aspect associated with this finding.

Enforcement: Title 10 CFR 20.1703(c) requires, in part, that the licensee implement and maintain a respiratory protection program that includes written procedures regarding the storage, issuance, maintenance of respiratory protection equipment. Contrary to this, as of January 16, 2008, the licensee failed to maintain procedures regarding the charging and proper maintenance of PAPR batteries. Because the failure to comply with 10 CFR 20.1703(c) was of very low safety significance and was entered into the licensee's corrective action program as CR-PLP-2007-04149 and CR-PLP-2008-00229, the violation is being treated as an NCV, (NCV 05000255/2008002-04: Failure to Maintain Procedures for the Maintenance of PAPR Batteries) consistent with Section VI.A of the NRC Enforcement Policy. Inadequate Engineering Controls

(2) Inadequate Engineering Controls

Introduction: A Green self-revealed finding of very low safety significance and associated NCV of 10 CFR 20.1701 was identified for failure to use, to the extent practical, process or other engineering controls to control the concentration of radioactive material in air.

Description: The licensee experienced instances of elevated airborne radioactivity in the Containment Building during the Fall 2007 refueling outage (1R19). The cause for these conditions was attributed to known fuel element failures identified early in the operating cycle.

On September 9, 2007, the licensee shut down the reactor for commencement of the planned refueling outage. The licensee monitored parameters of the reactor coolant system during the shutdown/cool-down process, including concentrations of key radionuclides. Radioactive noble gases were released to the containment atmosphere when the pressurizer manway was opened to support scheduled work. That activity created a short term condition where workers had difficulty leaving the radiologically controlled area (RCA) due to radioactive noble gases that would cling to the modesty clothing of the workers. Approximately 24 hours later, the licensee opened the steam generator manways to support scheduled work, which released more radioactive noble gases and later radioactive iodine to the containment atmosphere. When this event occurred, the licensee assessed the concentration of radioactive iodine in containment and assessed the impact on internal dose to workers. Additionally, the licensee expected that the installed engineering controls, which consisted of a charcoal filtered ventilation system, would remove the radioactive iodine from the atmosphere.

The duration of the elevated airborne radioactive iodine was much longer than anticipated by the licensee. The licensee's root cause evaluation determined that the charcoal media in the installed filtration system was depleted before the system was placed in service or shortly after the radioactive iodine was released to the containment atmosphere, thereby rendering the installed engineering controls ineffective. Prior to the outage, the licensee had elected not to replace the charcoal media within the installed plant equipment at the beginning of this refueling outage (1R19), as was performed during previous refueling outages. That decision was made after reviewing the results of a charcoal sample that was analyzed from the end of the previous refueling outage (1R18).

After the steam generator manways were removed, a local air filtration system was placed in service as prescribed during the ALARA planning process. The filtration system was a high-efficiency particulate air filter and a charcoal bank to remove radioactive iodine. The filter system was intended to draw air from the steam generator and into the plant removal system. However, the system components were installed backwards on the "A" steam generator. Instead of removing the radioactivity from the steam generator, the system effectively pushed unfiltered air out of the steam generator and into the containment atmosphere that created a localized increase in airborne radioactivity.

The prolonged, elevated airborne conditions that resulted from the exhaustion of the installed plant charcoal filtration units and the misalignment of the local high efficiency particulate air unit resulted in extended delays for workers as they attempted to leave the RCA and attributed to small but measurable intakes of radioactive iodine (I-131) to several hundred workers during 1R19. The licensee performed an assessment of each worker's internal dose and verified that all doses were well below regulatory limits. The licensee was considering various actions to prevent recurrence during future outages based on root cause evaluation recommendations.

Analysis: The inspectors determined that this finding was a performance deficiency because the licensee failed to meet the requirements contained in Subpart H of 10 CFR Part 20 and because the deficiency was reasonably within the licensee's ability to foresee and correct. The finding was more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of protecting worker health and safety from exposure to radiation, in that not implementing adequate engineering controls resulted in unplanned exposures to radioactive material. The finding was assessed using the Occupational Radiation Safety SDP and was determined to be of very low safety significance (Green) because it was not an ALARA planning issue, there was no overexposure or potential for overexposure, and the licensee's ability to assess dose was not compromised.

As described above, the engineering controls that were planned to be used to control the concentration of radioactive material in air were either depleted soon after being placed in service or installed improperly. Consequently, the cause of this deficiency had a cross-cutting aspect in the area of Human Performance related to work control. Specifically, the licensee failed to plan and coordinate work activities with planned contingencies and compensatory actions. (H.3(a))

Enforcement: Title 10 CFR 20.1701 requires that licensees use, to the extent practical, process or other engineering controls (e. g., containment, decontamination, or ventilation) to control the concentration of radioactive material in air. Contrary to this, between September 10-12, 2007, the licensee failed to implement effective engineering controls to control the concentration of radioactive material in air. Because the failure to comply with 10 CFR 20.1701 was of very low safety significance and has been entered into the licensee's corrective action program as CR-PLP-2007-04002, the violation is being treated as an NCV (NCV 05000255/2008002-05: Failure to Use, to the Extent Practical, Process or Other Engineering Controls to Control the Concentration of Radioactive Material in Air) consistent with Section VI.A of the NRC Enforcement Policy.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed corrective action reports related to access controls and high radiation area radiological incidents (non-Performance Indicators identified by the licensee in high radiation areas <1R/hr). Staff members were interviewed and corrective action documents were reviewed to verify that follow-up activities were being conducted in an effective and timely manner commensurate with their importance to safety and risk based on the following:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of NCVs tracked in the corrective action system; and
- Implementation/consideration of risk significant operational experience feedback.

This inspection constituted one sample as defined by Inspection Procedure 71121.01-5.

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

Radiological work in high radiation work areas having significant dose rate gradients was reviewed to evaluate the application of dosimetry to effectively monitor exposure to personnel and to assess the adequacy of licensee controls.

This inspection constituted one sample as defined by Inspection Procedure 71121.01-5.

b. Unresolved Item (URI)

The inspectors identified an URI concerning events that occurred on October 4, 2007, when three contract workers received electronic dosimeter dose rate alarms when they disassembled tools used for fuel reconstitution on the 649' level of the Auxiliary Building near the spent fuel pool. Radiation surveys performed after stainless steel inserts were placed in a box identified gamma dose rates greater than 100 millirem/hour and highly elevated beta radiation levels. At the time of the inspection, the licensee had not completed an evaluation of the radiological hazards of the work performed. As a result, the shallow dose for workers involved in the work evolution was unknown. Similarly, the inspectors could not evaluate the consequence of the apparent improper radiological posting for the area. Therefore, this issue remains under review by the NRC and is categorized as an URI (URI 05000255/2008002-06).

2OS2 ALARA Planning and Controls (71121.02)

.1 Radiological Work Planning

a. Inspection Scope

The inspectors evaluated the licensee's list of work activities ranked by estimated exposure that were in progress and reviewed the following work activities of highest exposure significance:

- RWP 754; Refuel Project – Reactor Vessel Disassembly;
- RWP 781; GSI-191 Project – CV-3001 and CV-3002 Replacement;
- RWP 765; Chemistry and Radiation Protection;
- RWP 823; Valve Repair Activities in Containment; and
- RWP 756; Refuel Project – Reactor Cavity Decontamination.

This inspection constituted one sample as defined by Inspection Procedure 71121.02-5.

For these activities, the inspectors reviewed the ALARA work activity evaluations, exposure estimates, and exposure mitigation requirements. The inspectors assessed whether procedures and engineering and work controls were based on sound radiation protection principles. The inspectors also evaluated whether the licensee had reasonably grouped the radiological work into work activities, based on historical precedence, industry norms, and/or special circumstances.

This inspection constituted one sample as defined by Inspection Procedure 71121.02-5.

The inspectors compared the results achieved including dose rate reductions and person-rem used with the intended dose established in the licensee's ALARA planning for the following work activities. Reasons for inconsistencies between intended and actual work activity doses were reviewed.

- RWP 754; Refuel Project – Reactor Vessel Disassembly;
- RWP 781; GSI-191 Project – CV-3001 and CV-3002 Replacement;
- RWP 765; Chemistry and Radiation Protection;
- RWP 823; Valve Repair Activities in Containment; and
- RWP 756; Refuel Project – Reactor Cavity Decontamination.

This inspection constituted one sample as defined by Inspection Procedure 71121.02-5.

The inspectors compared the person-hour estimates, provided by maintenance planning and other groups to the radiation protection group, with the actual work activity time requirements in order to evaluate the accuracy of these time estimates.

This inspection constituted one optional sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The licensee's process for adjusting exposure estimates or re-planning work, when unexpected changes in scope, emergent work or higher than anticipated radiation levels were encountered, was evaluated. This included assessing whether adjustments to estimated exposure (intended dose) were based on sound radiation protection and ALARA principles and not adjusted to account for failures to control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process.

This inspection constituted one sample as defined by Inspection Procedure 71121.02-5.

The licensee's exposure tracking system was evaluated to assess whether the level of exposure tracking detail, exposure report timeliness, and exposure report distribution was sufficient to support control of collective exposures. The RWPs were reviewed to assess whether they covered too many work activities to allow work activity specific exposure trends to be detected and controlled. During the conduct of exposure significant work, the inspectors evaluated if licensee management was aware of the exposure status of the work and would intervene if exposure trends increased beyond exposure estimates.

This inspection constituted one optional sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.3 Job Site Inspections and ALARA Control

a. Inspection Scope

Exposures of individuals from selected work groups were reviewed to evaluate any significant exposure variations which could exist among workers and to assess whether these significant exposure variations were the result of worker job skill differences or whether certain workers received higher doses because of poor ALARA work practices.

This inspection constituted one optional sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolution

a. Inspection Scope

The inspectors reviewed the licensee's self-assessments, audits, and Special Reports related to the ALARA program since the last inspection. The inspectors evaluated whether the licensee's overall audit program's scope and frequency met the requirements of 10 CFR 20.1101(c).

This inspection constituted one sample as defined by Inspection Procedure 71121.02-5.

The inspectors assessed whether identified problems were entered into the corrective action program for resolution, and that they had been properly characterized, prioritized, and resolved. This included dose significant post-job (work activity) reviews and post-outage ALARA report critiques of exposure performance.

This inspection constituted one optional sample as defined by Inspection Procedure 71121.02-5

Corrective action reports related to the ALARA program were reviewed and staff members were interviewed to assess whether follow-up activities had been conducted in an effective and timely manner commensurate with their importance to safety and risk using the following criteria:

- Initial problem identification, characterization, and tracking;
- Disposition of operability/reportability issues;
- Evaluation of safety significance/risk and priority for resolution;
- Identification of repetitive problems;
- Identification of contributing causes;
- Identification and implementation of effective corrective actions;
- Resolution of NCVs tracked in the corrective action system; and
- Implementation/consideration of risk significant operational experience feedback.

This inspection constituted one optional sample as defined by Inspection Procedure 71121.02-5.

The licensee's corrective action program was also reviewed to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed.

This inspection constituted one sample as defined by Inspection Procedure 71121.02-5.

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety [PS]

2PS3 Radiological Environmental Monitoring Program and Radioactive Material Control Program (71122.03)

.1 Unrestricted Release of Material from the Radiologically Controlled Area

a. Inspection Scope

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

This inspection does not meet the requirements to be counted as a sample as defined by Inspection Procedure 71122.03-5.

b. Findings

Introduction: A Green self-revealed finding of very low safety significance and associated NCV of 10 CFR 20.1501 was identified for the failure to conduct an adequate radiological evaluation in the form of surveys of contaminated workers.

Description: On January 17, 2008, the NRC notified the licensee that radioactive material was identified on workers entering another NRC licensed facility. The workers indicated that they had last been employed at the Palisades refueling outage (1R19) in September 2007. That licensed facility identified six pairs of footwear and other personal items with contamination levels between 6,000 and 20,000 disintegrations per minute. Subsequent analysis identified that the contamination was iodine-131, a radionuclide with an eight-day half life, and was linked to work activities at the Palisades site. The affected materials were confiscated by the other licensee after identification.

Prior to the release of the workers from the site, Palisades' staff had also identified two occurrences of inadequate surveys that were performed during the refueling outage that had resulted in the inadvertent release of licensed radioactive material from the restricted area. The incidents occurred approximately one week before the workers left Palisades to work at the other NRC licensed facility (described above). The immediate corrective actions taken by the licensee for these two events included communications to all radiation protection technicians that reinforced procedural compliance and the proper survey techniques for the release of individuals alarming contamination monitors. Additionally, a radiation protection supervisor was assigned (dayshift and nightshift) to provide additional oversight at access control.

As described in Section 2OS1.2, the licensee experienced elevated airborne radioactivity during the Fall 2007 refueling outage (1R19). The elevated airborne conditions resulted in low level intakes of radioactive material for numerous workers. Since the personal contamination monitors at the control points were not capable of differentiating any external contamination from the radioiodine intakes that caused them to alarm, the licensee relied on hand frisking to release the individuals and their personal items. The workers undergarments, shoes and socks were not independently surveyed and the licensee assumed that internal deposition of radioactive material was the only cause of the personal contamination monitor alarms. The requirement to perform manual surveys resulted in delays for workers as they attempted to leave the RCA and resulted in hundreds of worker being surveyed by radiation protection technicians using a pancake probe survey instrument, a technique also known as a hand frisk. The additional oversight provided by radiation protection supervisors was not fully effective because it did not provide adequate quality control that was warranted for the large number of personnel affected by the elevated airborne radioactivity. Consequently, contaminated personal items were released from the site undetected and were identified at another NRC licensed facility.

Analysis: The inspectors determined that this finding was a performance deficiency because licensees are required to adhere to the regulations of 10 CFR Part 20 and that the deficiency was reasonably within the licensee's ability to foresee and correct. The finding was more than minor because it impacted the program and process attribute of the Public Radiation Safety Cornerstone and affected the cornerstone objective to ensure adequate protection of public health and safety from exposure to radioactive material released into the public domain, in that inadequate surveys resulted in the failure to control radioactive material. The finding was assessed using Public Radiation Safety SDP and was determined to be of very low safety significance (Green). The finding was not a transportation issue, and the radioactive material found offsite was of low activity and would not have produced a dose to a member of the public in excess of 0.005 rem.

As described above, the actions required to survey the large number of workers that alarmed the personal contamination monitor overwhelmed the ability of the radiation protection staff to conduct effective hand frisks. Consequently, the cause of this deficiency had a cross-cutting aspect in the area of Human Performance. Specifically, the licensee failed to make risk-significant decisions using a systematic process, especially when faced with uncertain or unexpected plant conditions, to ensure safety is maintained. (H.1(a))

Enforcement: Title 10 CFR 20.1501 requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present. Pursuant to 10 CFR 20.1003, survey means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation.

Contrary to these requirements, on various dates in September 2007, the licensee did not perform adequate surveys to assure compliance with 10 CFR 20.1802, which requires that the licensee control and maintain constant surveillance of licensed material that is in a controlled area or unrestricted areas and that is not in storage. Specifically, between September 21, 2007, and September 30, 2007, licensee surveys of workers were not adequate to control licensed material from inadvertently being carried by the workers outside of the controlled and restricted areas of the site. Because this finding is of very low safety significance and has been entered into the licensee's corrective action program (Condition Reports CR-PLP-2007-04338 and CR-PLP-2008-01180), this violation is being treated as an NCV (NCV 05000255/2008002-07, Failure to Control the Release of Radioactive Material), consistent with Section VI.A of the NRC Enforcement Policy.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Quarterly Data Submission

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the first quarter 2007 Performance Indicators for any obvious inconsistencies prior to its public release in accordance with Inspection Manual Chapter 0608, “Performance Indicator Program.”

This review was performed as part of the inspectors’ normal plant status activities and, as such, did not constitute a separate inspection sample.

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Scrams per 7000 Critical Hours Performance Indicator for the period from the first through the fourth quarter of 2007 to determine the accuracy of the Performance Indicator data reported during those periods, Performance Indicator definitions and guidance contained in Revision 5 of the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” were used. The inspectors reviewed the licensee’s operator narrative logs, condition reports, event reports and NRC Integrated Inspection reports to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the Performance Indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one sample for unplanned scrams per 7000 critical hours sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

.3 Mitigating System Performance Indicator (MSPI) HPSI

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI – HPSI for the period from the first through the fourth quarter of 2007 to determine the accuracy of the Performance Indicator data reported during those periods, Performance Indicator definitions and guidance contained in NEI Document 99-02 were used. The inspectors reviewed the licensee’s operator narrative logs, condition reports, event reports and NRC Integrated

Inspection reports to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the Performance Indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one MSPI – HPSI sample as defined by Inspection Procedure 71151.

.4 Unplanned Scrams With Complications

a. Inspection Scope

The inspectors sampled licensee submittals for the unplanned scrams with complications Performance Indicator for the period from the first through the fourth quarter of 2007 to determine the accuracy of the Performance Indicator data reported during those periods, Performance Indicator definitions and guidance contained in NEI Document 99-02 were used. The inspectors reviewed the licensee's operator narrative logs, condition reports, event reports and NRC Integrated Inspection reports to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the Performance Indicator data collected or transmitted for this indicator and none were identified. Specific documents reviewed are described in the Appendix to this report.

This inspection constitutes one unplanned scrams with complications sample as defined by Inspection Procedure 71151.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of items Entered Into the Corrective Action Program

a. Inspection Scope

As part of the various baseline inspection procedures discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program 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: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that 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.

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.

.2 Review of Daily Corrective Action Documents

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 corrective action program. 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.

.3 Annual Sample: Motor Operated Valve (MOV) and Air Operated Valve (AOV) Performance

a. Inspection Scope

The inspectors completed one inspection sample regarding problem identification and resolution by conducting an in-depth review of automatic valve performance. The inspectors noted in the last year there had been several failures of air operated, safety-related valves. The valves included: the failure of CV-0737, C AFW pump flow control bypass to the A steam generator; CV-0703, the B feed regulating valve; and CV-3047, C Safety Injection Tank pressure control valve. All the failures resulted in the valve failing in a position which was not the fail-safe position. The inspector's scope was to determine if there was a common cause to the failures. The inspectors reviewed work orders, condition reports and cause evaluations. In addition, the inspectors interviewed site personnel familiar with the issues and conducted independent condition report searches to determine if other valves were experiencing similar failures. Finally, the inspectors assessed the licensee's resolution of the individual and collective issue for automatic valve performance issues.

b. Assessment and Observations

The failure assessment by the licensee for two valves, CV-3047 and CV-0737, did determine the direct material cause, but could not determine how the material condition occurred. For CV-3047 a gasket on the air relay diaphragm was found torn, but since the pilot had been replaced less than four years ago, it did not appear to be age related. For CV-0737, the positioner had a nut/lock washer which had backed out, but there was no indication of how this occurred. Since the valves had work performed on them, human performance and work practices could not be ruled out. The corrective actions were reasonable based on the limited information available.

The inspector's review of the apparent cause evaluation for CV-0703 noted several deficiencies. The inspector's review was done on the approved evaluation in the beginning of February. Although the apparent cause evaluation was completed, it had not been reviewed by the Corrective Action Review Board even though the transient occurred during start-up operation on October 21, 2007. While the plant was in mode 2 at about 3 percent power following a refueling outage, the operations staff attempted to transfer from auxiliary feedwater to main feedwater. When the B feed regulating valve, CV-0703, was un-isolated, primary coolant temperature dropped and Steam Generator level began to rise. Although CV-0703 was believed to be closed and control room indications showed the valve to be closed, it was partially open. Temperature dropped to within 0.3 F of the minimum temperature for critical operations required by TS as Steam Generator level rose to 86 percent. The operations staff backed out of the procedure and took action to repair the valve. The cause evaluation did not address how the post maintenance testing missed the fact the valve was 16 percent open and was returned to operations with the valve not in its safety position. The evaluation did not assess operations' performance to determine if they responded adequately or if operations procedures were adequate for the event. The cause evaluation did not address the equipment performance of other mitigating systems. Although the logs indicate that Steam Generator level reached 86 percent, the high level override (a non-safety grade feature) which should send a full closed actuation signal to the feed regulating valves was not received at 84.7 percent. The inspectors interviewed shift personnel and their assessment was that there is some inaccuracy in the plant computer points or the actuation signal. In addition, the cause evaluation stated there was no impact on the maintenance rule function. This incorrect assessment is discussed in Section 1R12.

The cause evaluation determined that after testing and adjustments of the AOV, the locknuts which secured the bias spring were tightened. This caused bias spring movement and a shift of the zero adjustment. This information is not in any work order, procedure or vendor manual. The cause evaluation concluded it is fundamental knowledge to tighten the locknuts before final acceptance testing; and focused on the experience level of the augmented work force. The inspectors drew a different conclusion: since the adjustment is critical for the proper adjustment of the AOV in the 'as left' condition, it should be captured in a quality document to ensure it is performed correctly. This feedback was provided to site personnel; and corrective actions to address these issues are planned to be done in CR-PLP-2007-5375. Corrective Action Review Board approved the apparent cause evaluation on March 25, over five months after the event. The inspectors concluded the cause, as written, did not adequately address the issues associated with the plant transient.

The inspectors reviewed CRs to look for degrading trends for AOV and MOVs. The inspectors noted a CR in December 2007 for MO-3199, Low Pressure Safety Injection Pump A inlet valve crosstie. The valve is experiencing degradation due to internal guide wear. The inspectors noted some similarity with a previous failure of CV-3070 in that loads on the valve could impact valve performance. For MO-3199, the licensee placed lead shielding on the body to bonnet flange to reduce dose rate with the valve that is mounted horizontally. The lead shielding added weight to the valve. No specific evaluation was done to determine if that weight could impact valve operation. The licensee closed the CR to WO294518 (which would be completed during a outage when all the fuel in core is removed). Although currently meeting acceptance criteria, the inspectors were concerned that future valve diagnostics and repair were not scheduled for at least another four years and that further degradation may not be detected until the valve could no longer be considered operable. After discussions with the inspectors, the licensee wrote CR 2008-01054 to establish additional monitoring actions. Based on current valve performance, the inspectors concluded the issues are minor. However, monitoring actions are needed to ensure margins remain adequate.

Overall, the inspectors found no common thread which impacted these valves, but did conclude that site personnel performing maintenance contributed to some of the deficiencies.

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

.1 Plant trip due to Loss of a Main Feed Pump

a. Inspection Scope

The inspectors reviewed the plant's response to a manual plant trip on January 13, 2008 caused by a loss of a Main feed Pump at 100 percent power. The inspectors observed post trip action in the control room and reviewed the licensee's post-scram report. This inspection constitutes one sample as defined in Inspection Procedure 71153-05.

b. Findings

Introduction: A Green self-revealed finding occurred on January 13 when the 'B' Main Feed Pump failed. The failure occurred due to improper maintenance on the lube oil pump associated with the Main Feed Pump that resulted in a loss of lube oil flow and trip of the Main Feed Pump.

Description: On January 13, with the plant at 100 percent power, the 'B' Main Feed Pump tripped due to a loss of lube oil pressure. In accordance with Off- Normal Procedure ONP-12, Loss of main Feedwater, operators manually tripped the reactor. Following the trip, the licensee formed an incident response team to determine what caused the feed pump trip. The team identified that the drive coupling between the shaft driven lube oil pump and the feed pump failed causing a loss of lube oil pressure and subsequent Main Feed Pump trip. A root cause team determined that following maintenance in the fall 2007 outage, the pump coupling had been reassembled with insufficient engagement between the shaft coupling hub and outer sleeve. The lack of engagement resulted in rapid wear of the hub and coupling splines eventually leading to the coupling's failure.

The root cause team determined the improper reassembly resulted from use of an improper key between the drive shaft and the hub. The proper key includes a foot to limit the distance the hub can be slid up the shaft. The work instructions used for reassembly of the pump lacked sufficient detail to ensure the proper key was used. In addition, the key in use had either been modified during previous pump maintenance to remove the foot or a key without a foot was substituted for the correct key.

Analysis: The inspectors determined the failure to use the proper key in the Main Feed Pump was a performance deficiency that warranted a safety significance determination. The inspectors concluded that the finding was more than minor in accordance with Inspection Manual Chapter 0609 because the finding is associated with the reactor safety cornerstone objective of reducing the likelihood of an initiating event. Specifically, the improper pump assembly led to a partial loss of feed and subsequent plant trip. The inspectors reviewed the finding in accordance with Inspection Manual Chapter 0612. In accordance with the phase one screening checklist, because the finding did not affect a mitigating system in addition to being a transient initiator, the finding was of very low safety significance, i.e. Green. Since the finding occurred because the documentation of the key lacked sufficient detail to ensure proper assembly, the finding included a cross-cutting aspect in the area of Human Performance, Resources, Complete and Accurate Documentation (H.2(c)).

Enforcement: The finding does not represent a violation of NRC requirements. However, since it represents a failure to meet a self imposed requirement, the inspectors concluded the deficiency constituted a finding consistent with Section VI.A.1 of the NRC Enforcement Policy. Specifically, FP-WM-PLA-01, Work Order planning process, stipulates that task instructions should match the complexity of the activity commensurate with the qualifications of the workers. Contrary to this, the task instruction did not include sufficient detail to properly reassemble the Main Feed Pump lube oil pump coupling. Therefore, this finding is identified as Finding (FIN) 05000255/2008002-08, Improper Main Feed Pump Coupling Assembly. This issue is in the licensee's corrective action program as CR-PLP-2008-0151.

.2 (Closed) Licensee Event Report (LER) 05000255/2008001-00: Reactor Protection System and Feedwater System Actuation

On January 13, 2008 with the plant in mode 1, main feedwater pump 1-B tripped unexpectedly. Operators, in accordance with abnormal operating procedures, manually tripped the plant. As expected for the transient, auxiliary feedwater actuated to recover Steam Generator levels. The immediate cause was low lube oil pressure. The low lube oil pressure resulted from a loss of the shaft driven lube oil pump. The cause was determined to be an incorrect shaft key which permitted partial engagement of the gearing which eventually wore such that the lube oil pump was no longer being driven. The assessment of plant response was described in the above section. One finding was identified, which is discussed above as FIN 05000255/2008002-08. No other safety concerns were identified. This LER is closed.

.3 (Closed) LER 05000255/2007010-00: TSs Action Requirements Not Met for Battery Cell Parameter Outside Allowable Limits

a. Inspection Scope

On December 27, 2007, during the performance of TS surveillance testing of the main station batteries, the float voltage of battery cell 43 on the right train station battery was below the allowable TS limit for this parameter. However, at the time of this discovery, the performer did not recognize that the as-found value fell below the specified TS battery cell limit. The procedure used included the correct acceptance criteria. On December 28, 2007, during review of the surveillance data and discussion with members of the electrical maintenance department who had performed the surveillance, an on-duty senior reactor operator recognized the low reading for battery cell 43. The delay of over 24 hours in recognizing that battery cell 43 float voltage was below the TS limit for this parameter resulted in not meeting the TS completion time for required actions in accordance with TS 3.8.6.A, TS 3.8.6.B, TS 3.8.4 B that were applicable from the initial discovery time. Finally a shutdown required in accordance TS 3.8.4C was not completed based on the battery performance. One finding was identified. This LER is closed.

b. Findings

Introduction: A self revealing NCV of TS 3.8.4 B and C was identified for failing to recognize that battery cell parameters were not within TS limits and for failing to take actions accordance with TS for an inoperable battery. Specifically, cell 43 of the right train safety-related battery (ED02,) was below TSs limits for individual cell voltage without recognition by the site staff. As a result, compensatory actions and a plant shutdown required by TSs were not completed as required.

Description: On December 27, 2007, during the performance of TS surveillance testing of the main station batteries, the float voltage of battery cell 43 on the right train station battery was below the allowable TS limit for this parameter. However, at the time of this discovery, the performer did not recognize that the as-found value fell below the specified TS battery cell limit. On December 28, 2007, during review of the surveillance data and discussion with members of the electrical maintenance department who had performed the surveillance, an on-duty senior reactor operator recognized the low reading for battery cell 43. The delay of over 24 hours in recognizing that battery cell 43 float voltage was below the TS limit for this parameter resulted in not meeting the TS completion time for required actions in accordance with TS 3.8.6.A and TS 3.8.6.B, that were applicable from the initial discovery time.

The required action to immediately declare the right train station battery inoperable was not met. Additionally, with the right train station battery inoperable, TS 3.8.4.B requires a verification that both the directly connected and cross-connected battery chargers are supplying power to the affected train with a completion time of two hours, and that the station battery be restored to operable status within 24 hours. This action was not completed in the two hours. With these required actions and associated completion times not met, the required actions of TS 3.8.4.C requiring Mode 3 entry in six hours was also not met.

The shift manager declared the battery inoperable after being informed of the condition of cell 43. The licensee completed the required actions of 3.8.4 B within two hours. The licensee replaced the cell and verified acceptable performance during a post maintenance test. The licensee determined the cause of the failure to recognize the surveillance failure was an inadequate pre-job brief.

Analysis: The inspectors determined that the failure of the site to initially recognize that battery cell 43 voltage was below the TS was within the licensee's ability to foresee and correct. The failure to take required actions in accordance with TSs was more than minor because the TS actions increase reliability of the Direct Current (DC) bus. Therefore, the finding impacted the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the objective to ensure availability, reliability and capability of the systems which respond to initiating events. The finding is of very low safety significance (Green), because the finding did not cause a loss of safety function for the right train battery. The finding includes a cross-cutting aspect in the area of human performance in that human error prevention techniques (H.4(a)), in this case a pre-job brief, were not effective in preventing the delay in notification of the senior reactor operators.

Enforcement: Technical Specification 3.8.4 Action B requires, in part, that in two hours an operable cross-connected and directly connected charger be connected to the affected DC bus when one power source battery is in operable. In addition TS 3.8.4 Action C requires the plant be placed in mode 3 in six hours when the required action and associated completion times are not met. Contrary to this, on December 28, 2008, with the right train battery (ED02) inoperable, both battery chargers were not placed in service in two hours; and the plant was not placed in mode 3 within six hours. Once the shift manager became aware of the status of the battery, the licensee completed the required actions. The failure to take actions required by TS is being treated as a non-cited violation (NCV), consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000255/2008002-09: Failure to Comply with TS 3.8.4 B and C. This issue is in the licensee's corrective action program as CR-PLP-2007-06496. The licensee replaced cell 43 for battery ED02.

.4 <u>(Closed) LER 05000255/2007009-00: Automatic Valve in Emergency Core Cooling System Inoperable in Excess of TS Requirements</u>

a. <u>Inspection Scope</u>

On November 26, 2007, CV-3047 (a normally closed, automatic valve which is opened periodically for safety injection tank operations; and closes on safety system actuation) had exceeded its stroke time to close during testing. Pending further troubleshooting, administrative controls were established with the intent to maintain CV-3047 closed. Subsequently, on December 18, 2007, investigation determined that CV-3047, although indicating closed, was not fully closed. Technical Specification Surveillance Requirements 3.5.2.2 and 3.5.2.5 require that each ECCS automatic valve in the flow path be verified to be in the correct position, and to actuate to the correct position, respectively. Since CV-3047 was not fully closed and would not fully close, it was incapable of meeting Surveillance Requirements 3.5.2.2 and 3.5.2.5. The plant was in this condition in excess of the TS completion time of 72 hours for one inoperable train. One finding as documented below was identified. This LER is closed.

b. <u>Findings</u>

<u>Introduction:</u> A self revealing NCV of TS 3.5.2 B and C was identified for the inability of an automatic valve in the ECCS, CV-3047, to reposition fully closed on an actuation signal. As a result, one train of ECCS was inoperable for longer than allowed by TSs.

<u>Description:</u> On November 26, 2007, CV-3047 (a normally closed, automatic valve which is opened periodically for safety injection tank operations; and which closes on safety system actuation) exceeded its stroke time to close during testing. CV-3047 is intended to close, along with other valves to ensure ECCS flow through the core is not bypassed in the event of a postulated loss of coolant accident. The valve was declared inoperable; and pending further troubleshooting, administrative controls were established with the intent to maintain CV-3047 closed in its safety position. The administrative action was to place a tag indicating the valve should not be opened. The site investigated possible actions to repair the valve, but believed that the radiation field was too high to repair the valve. They did not look at actions or activities to either verify the valve was closed locally or to verify no flow was occurring through the valve.

Subsequently, on December 18, 2007, during safety injection tank operations, investigation determined that CV-3047, although indicating closed, was not fully closed. TS Surveillance Requirements 3.5.2.2 and 3.5.2.5 require that each ECCS automatic valve in the flow path be verified to be in the correct position, and to actuate to the correct position, respectively. Since CV-3047 was not fully closed, it was incapable of meeting Surveillance Requirements 3.5.2.2 and 3.5.2.5. This rendered one train of ECCS inoperable. The licensee wrote a CR (CR-PLP-2007-06351) and manually isolated the flow path to comply with TS. The licensee repaired the valve, successfully retested it, and restored the valve to service. The time the valve was partially open between November 26 and December 18, 2007, about 23 days, exceeded TS requirements of 72 hours.

Analysis: The inspectors determined the failure to ensure the valve was closed was within the licensee's ability to foresee and correct. The failure to take required actions in accordance with TS was more than minor because the finding impacted the equipment performance attribute of the Mitigating Systems cornerstone and adversely affected the objective to ensure availability, reliability and capability of the systems which respond to initiating events. More flow would bypass than core with the valve approximately 18 percent open than if the valve had been fully closed. The licensee performed analyses to determine the ECCS flow with the valve partially open. The bypass flow would not have prevented the ECCS safety function from being maintained based on current plant analysis. Therefore the finding was considered to have very low safety significance (Green). The finding included a cross-cutting aspect in the area of human performance in that licensee did not adequately coordinate work activities to address the impact of actions needed to ensure the valve was closed when the valve was declared inoperable. The consideration of using cameras, surveys, alternate methods for ensuring the valve was closed was not followed through on by the site team to ensure adequate equipment performance. (H.3(b)).

Enforcement: Technical Specification Surveillance Requirements 3.5.2.2 and 3.5.2.5 require, in part, that each ECCS automatic valve in the flow path be verified to be in the correct position, and to actuate to the correct position, respectively. Surveillance Requirement 3.0.1 states, in part, failure to meet a surveillance, shall be failure to meet the LCO. LCO for TS 3.5.2 requires two ECCS trains operable. Technical Specification 3.5.2 Action B requires, in part, that with one ECCS train inoperable, the inoperable train be restored to operable in 72 hours. In addition TS 3.5.2 Action C requires the plant to be placed in mode 3 in six hours when the required action and associated completion times are not met. Contrary to this, on November 29, 2007, with one train of ECCS inoperable, due to the inability of CV-3047 to meet the above surveillances with the valve not in its correct position, the train was not restored to service in 72 hours nor was the plant placed in mode 3 in the required time. The failure to take actions required by TS is being treated as a Non-Cited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy and is identified as NCV 05000255/2008002-10: Failure to Comply with TS 3.5.2 B and C. This issue is in the licensee's corrective action program as CR-PLP-2007-06351. The licensee completed repairs to CV-3047.

4OA5 Other Activities

.1 Preoperational and Operational Testing of an Independent Spent Fuel Storage Installation (60854.1)

a. Inspection Scope

During this quarter, an inspection of the licensee's dry fuel storage dry run and loading of the first cask of the campaign was initiated. Inspection activities will continue into the second quarter. Results of the inspection will be included in next quarter's report (Inspection Report No. 05000255/2008003).

.2 Closure of Unresolved Items

Closure of URI 05000255/2007006-04: Internal Dose Assessment for O-ring Work

During the Fall 2007 refueling outage, an individual's respiratory protection equipment failed during the removal of the reactor head O-ring. That event resulted in an intake of radioactive material. However, the internal dose assessment was not complete, because the licensee had not assessed the dose impact form non-gamma emitting radionuclides. In accordance with plant procedures, samples (i.e., area contamination and in vitro bioassay) were collected and sent to a contracted off-site facility to perform analysis for those difficult to detect radionuclides. The results from the analysis were not available during that inspection. Additionally, the causes for the respiratory equipment failure were still under evaluation.

The licensee conducted additional evaluations and analysis of the event and performed a dose evaluation from the intake of radionuclides. The inspectors determined that the dose evaluation was adequate and that doses were below NRC limits. The initiating event of failed PAPR batteries was reviewed as described in Section 2OS1.2 and resulted in a finding of very low safety significance and an NCV. The URI is closed.

(Closed) URI 05000255/2007006-05: Increased Airborne Radioactivity In Containment

During the Fall 2007 refueling outage, airborne radioactivity areas were created within the containment building. Increased levels of noble gas were identified after the pressurizer manway was removed, and the levels increased again after the steam generator manways were removed, as part of the work that was scheduled during the refueling outage. Increased levels of iodine-131 were identified after the reactor head was lifted to support the refueling outage. The increased airborne radioactivity levels caused small, but measurable, intakes of iodine-131 to several hundred workers during the refueling outage. At the time of that inspection, the events were still under review with respect to the causes of the events, the extent of the personnel intakes, the adequacy of pre-job planning, and the adequacy of contingency actions to mitigate the conditions before allowing work to continue in the affected areas.

The licensee performed a root cause evaluation of the incident and also evaluated the doses to the workers. As described in Section 2OS1.2, a finding of very low safety significance (Green) and an NCV were identified for the failure to implement adequate engineering controls to reduce the levels of airborne radioactivity. However, the doses to the workers were maintained below NRC requirements. The URI is closed.

.3 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

During the inspection period, the inspectors conducted the following observations of security force personnel and activities to ensure that the activities were consistent with licensee security procedures and regulatory requirements relating to nuclear plant security. These observations took place during both normal and off-normal plant working hours.

- Multiple tours of operations within the security alarm stations;
- Tours of selected security officer response posts;
- Direct observation of personnel entry screening operations within the plant's Main Access Facility;
- Security force shift turnover activities; and
- These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

These quarterly resident inspector observations of security force personnel and activities did not constitute any additional inspection samples. Rather, they were considered an integral part of the inspectors' normal plant status review and inspection activities.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On April 23, the inspectors presented the inspection results to C. Schwarz and other members of the licensee staff. The licensee acknowledged the issues presented. The inspector asked the licensee whether any materials examined during the inspection should be considered proprietary. Proprietary information was identified and will be handled in accordance with established procedures.

.2 Interim Exit Meetings

An interim exit meeting was conducted for the licensed operator requalification training biennial written examination and annual operating test results with the Site Vice President, Mr. C. Schwarz, on February 8, 2008.

An interim exit meeting was conducted for the annual operating test results with the Superintendent Operations Training, Mr. T. Davis, via telephone on February 20, 2008.

An interim exit meeting was conducted for the occupational radiation safety program for access to radiologically significant areas and ALARA planning and controls with Mr. C. Schwarz on January 17, 2008, and Mr. T. Kirwin on February 22, 2008.

An interim exit meeting was conducted for the public radiation safety program for radioactive material control program with Ms. L. Lahti on March 21, 2008.

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green) were identified by the licensee and are violations of NRC requirements which meet the criteria of Section VI of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as NCVs.

- Technical Specification 5.7.1 requires areas with dose rates greater than 100 milirem/hour to be posted and controlled as a High Radiation Area. Contrary to this, on September 20, 2007, and other dates, the high radiation area posting and barricade was found altered and ineffective on the 590' elevation of containment. This was identified in the licensee's corrective action program as CR-PLP-2007-04236. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised.

- Technical Specification 5.7.2 requires areas with dose rates greater than 1000 millirem/hour to be posted and controlled to prevent inadvertent entry. Contrary to this, on September 22, 2007, a steam generator platform worker left the work area before the steam generator hand hole covers were in place. This configuration allowed inadvertent access to an area where rates exceeded 1000 millirem/hour. This was identified in the licensee's corrective action program as CR-PLP-2007-04304 and was reported as a Performance Indicator occurrence. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

C. Schwarz, Site Vice President
S. Bell, Radiation Protection Dosimetry Program Owner
L. Blocker, Operations Manager
J. Broschak, Engineering Director
N. Brott, Emergency Preparedness Coordinator
T. Davis, Operations Prequalification Supervisor
B. Dotson, Regulatory Compliance
R. Farrell, Radiation Protection Manager
M. Ginzel, Radiation Protection Manager
P. Johnson, Safety Manager
T. Kirwin, Plant General Manager
L. Lahti, Licensing Manager
D. Malone, Regulatory Affairs
B. Nixon, Assistant Operations Manager
M. Richey, Acting Plant General Manager
P. Schmidt, Simulator Training Supervisor
G. Sleeper, Assistant Operations Manager
G. Sturm, Radiation Protection Planner
K. Smith, Quality Assurance Manager
B. Smoot, Radiation Protection Supervisor
J. Walker, Operations
T. Watson, Operations Prequalification Training Instructor
P. Williams, RP Supervisor – Outage ALARA Planner

Nuclear Regulatory Commission

D. McNeil, Senior Operations Engineer
R. Walton, Operations Engineer

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000255/2008002-01	NCV	Failure to Ensure Fire Door Was Closed (Section 1R05)
05000255/2008002-02	NCV	Failure to Monitor the Feedwater System Under 10 CFR 50.65a(1) (Section 1R12)
05000255/2008002-03	NCV	Inadequate General Operating Procedure for Mode Transition (Section 1R20)
05000255/2008002-04	NCV	Failure to Maintain Procedures for the Maintenance of PAPR Batteries (Section 2OS1)
05000255/2008002-05	NCV	Failure to Use, to the Extent Practical, Process or Other Engineering Controls to Control the Concentration of Radioactive Material in Air (Section 2OS1)
05000255/2008002-06	URI	Failures to evaluate the shallow (skin) dose to three workers involved in tool disassembly and failure to barricade and conspicuously post each entryway to a high radiation area (Section 2OS1)
05000255/2008002-07	NCV	Failure to Control the Release of Radioactive Material (Section 2PS3)
05000255/2008002-08	FIN	Main Feed Pump Trip due to Inadequate Configuration (Section 4OA3)
05000255/2008002-09	NCV	Failure to Comply with TS 3.8.4 B and C (Section 4OA3)
05000255/2008002-10	NCV	Failure to Comply with TS 3.5.2 B and C (Section 4OA3)

Closed

05000255/2008002-01	NCV	Failure to Ensure Fire Door Was Closed (Section 1R05)
05000255/2008002-02	NCV	Failure to Monitor the Feedwater System Under 10 CFR 50.65a(1) (Section 1R12)
05000255/2008002-03	NCV	Inadequate General Operating Procedure for Mode Transition (Section 1R20)
05000255/2008002-04	NCV	Failure to Maintain Procedures for the Maintenance of PAPR batteries (Section 2OS1)
05000255/2008002-05	NCV	Failure to Use, to the Extent Practical, Process or Other Engineering Controls to Control the Concentration of Radioactive Material in Air (Section 2OS1)
05000255/2008002-07	NCV	Failure to Control the Release of Radioactive Material (Section 2PS3)
05000255/2008002-08	FIN	Main Feed Pump trip due to Inadequate Configuration (Section 4OA3)
05000255/2008002-09	NCV	Failure to Comply with TS 3.8.4 B and C (Section 4OA3)
05000255/2008002-10	NCV	Failure to Comply with TS 3.5.2 B and C (Section 4OA3)
05000255/20070009	LER	Automatic Valve in Emergency Core Cooling System Inoperable in Excess of TS Requirements (Section 4OA3)
05000255/20070010	LER	Technical Specifications Action Requirements Not met for battery Cell Parameter Outside Allowable Limits (Section 4OA3)

05000255/20080001	LER	Reactor Protection System and Feedwater System Actuation (Section 4OA3)
05000255/2007006-04	URI	Internal Dose Assessment for O-ring Work (Section 4OA5)
05000255/2007006-05	URI	Increased airborne radioactivity in containment (Section 4OA5)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection

- CR-PLP-2008-00655, Service Water Pump 7C Basket Strainer DP HI, February 10, 2008
- CR-PLP-2008-00651, Service Water Pump 7A Basket Strainer DP HI, February 10, 2008
- CR-PLP-2008-00687, Exciter Air Cooler became Partially Plugged during Turbulent Lake Conditions, February 10, 2008

1R04 Equipment Alignment

- SOP-12, Feedwater System, Revision 52
- Attachment 14, Auxiliary Feedwater System checklist 12.5, Revision 52
- MO-29, Engineered Safety System Alignment, Revision 35
- SOP-3, Safety Injection and Shutdown cooling System, Revision 74

1R05 Fire Protection

- List of Changes and Response to Appendix A to Branch Technical Position APCSB 9.5-1 and Regulatory Guide 1.78 and 1.101, August 24, 1996
- Fire Hazards Analysis, Palisades Plant, Revision 7
- CR-PLP-2008-00075, Fire Door 71 not latched closed, January 8, 2008

1R11 Licensed Operator Requalification Program

- Current List and Summary of Simulator Work Requests, February 2008
- FP-T-SAT-81, Simulator Testing and Documentation, November 3, 2006
- EM-04-24, Palisades Critical Prediction and Critical Approach, Revision 7
- Simulator Written Examinations 1 through 5, Administered January – February 2007
- Steady State Simulator Test SS-01, November 21, 2006
- Simulator Transient Tests, (various), October – December 2006
- Simulator Core Performance Test
- Simulator Normal Evolutions Test N-02, Unit Startup from Hot Standby
- Simulator Operating Test RT&R, Real Time and Repeatability Test, November 6, 2007
- Annual Requalification Operating Tests, 6 Crews, various dates
- Biennial Written Examinations 5 Examinations, various dates
- Nuclear Oversight Observation Report #2006-002-8-026, May 17-19, 2006
- Conduct of On the Job Training and Task Performance Evaluation Report # 2006-004-8-013
- Simulator Dynamic Evaluations Report #2007-001-8-001, December 26, 2006
- Training and Qualification, January 2, 2007 – January 19, 2007
- Oversight Observation Checklist 02C-PAL-2007-0151, September 5, 2007 – Observation of Operations Just in Time Training – Task Training for RFO 19
- Oversight Observation Checklist 02C-PAL-2007-0304, 09/29/2007 – Approach to Critical Following RFO 19

- Oversight Observation Checklist 02C-PAL-2007-0305, September 29, 2007 – Low Power Physics Testing following RFO 19
- Licensed Operator Examination Security, Revision 6, September 26, 2007
- Palisades Nuclear Plant Nuclear Training Procedure PNT 12.0
- Control of Time Critical Operator Actions, FP-OP-CTC-01, Revision 0, September 29, 2006
- Remediation Training Form QF-1040-04, Revision 4 (FP-T-SAT-40) 3 Individual Operators, various dates
- Conduct of Operations EN-OP-115, Revision 5
- Licensed Operator Requalification Program Examinations FP-T-SAT-73, Revision 2, April 9, 2007
- NRC Operator License Application and Renewal Requirements FP-T-SAT-74, Revision 1, January 12, 2007
- JPM and Simulator Exercise Guide Development FP-T-SAT-75, Revision 0, December 27, 2006
- Requalification Training Attendance Roster, various dates
- Simulator evaluation for January 18, 2008

1R12 Maintenance Effectiveness

- EM-25, Maintenance Rule Program, Revision 6
- EM-20, Performance Monitoring, Revision 12
- System health report July 2007, feedwater system
- CR-PLP-2008-00393, CA-1, evaluate FWS/MFW for (a)(1) status, February 25, 2008
- CR-PLP-2007-5375, apparent cause evaluation CV-0703 control Problem, November 10, 2007
- CR-PLP-2008-00562, Maintenance Rule Evaluation Not Completed, February 6, 2008
- MSM-M-57, Universal Diagnostics System Operating Procedure, Revision 7
- CR-PLP-2007-03424, MSM-M-57 Operating procedure on CV-0511, August 23, 2007
- Executive Summary for MFW System, February 6, 2008
- CR-PLP-2007-05380, Troubleshooting CV-0703 found POC-0703 zero shifted by 2 psi, October 21, 2008
- QO-1, Main Feedwater Regulating Valves Inservice Stroke test, Revision 7
- ONP-10, Excessive Feedwater Increase, Revision 6
- Maintenance Trend report, Fourth Quarter 2007
- CR-PLP-2007-5918, Maintenance Rework, Revision 0
- SOP-12, feedwater System, Revision 52
- EGAD-EP-10, Maintenance Rule Scoping Documents, Revision 4
- System Health Report, Primary Coolant System, March 2008
- DBD-2.04, Primary Coolant System, Revision. 6
- CR-PLP-2007-05256, Leakage at Primary Coolant Pump P-50C Casing Flange, October 15, 2007

1R13 Maintenance Risk Assessments and Emergent Work Control

- OPS risk assessment for 1/16/2008, EOOS
- RE-135, Battery Charger No. 3 Performance test completed June 28, 2006
- OPS risk assessment for 2/5/2008, EOOS
- Weekly schedule January 31, 2008
- Weekly schedule March 3, 2008
- Weekly Schedule March 11, 2008

1R15 Operability Evaluations

- CR-PLP-2008-00371 – declining voltage ED02 cell 23, January 25, 2008
- QE-35, ED-01 and ED-02 Battery Checks – Quarterly, Revision 6
- CR-PLP-2005-02144, Fuel Handling Procedure Appears to Allow TS Prohibited Action, April 18, 2005
- CR-PLP-2008-00614, VC-11 Chiller failed to Start, February 7, 2008
- ACE-003554, Fuel handling procedure GOP-1, Revision 0
- GOP-11, Refueling Operations and Fuel Handling, Revision 42
- CR-PLP-2008-00822, Broken Valve keeper Found on Cylinder 2L of 1-2 EDG
- CR-PLP-2008-01131, Main Steam Relief Flange connections nuts not tightened, March 10, 2008
- PLP-RPT-08-00001, Entergy Calculation Evaluation of RV-0719, E-50B Main Steam Safety Valve, Flange Bolting, Revision 0
- CR-PLP-2008-01382, CV-3006, SDC Heat Exchanger bypass, ACME Screw Found Flush with Stem Connector, March 26, 2008

1R19 Post Maintenance Testing

- QO-014, Inservice Test Procedure- Service Water Pumps, Revision. 14
- MO-7A-1, 1-1 EDG surveillance testing performed 1/7/08
- WO00329794, CV-3047 SIT T-82C PCV Failed Max Closure Stroke Time, January 2008
- CR-PLP-2008-00708, EA-GOTHIC-AFW-02, AFW pump room heat-up calc., February 12, 2008
- T-186, Auxiliary Feedwater Turbine K-8 Overspeed Trip Test and Governor Setting, Revision 14
- RO-145, Comprehensive Pump Test Procedure Auxiliary Feedwater P-8A, P-8B, and P8-C, February 14, 2008
- NRC Information Notice 86-14 Overspeed Trips of AFW Turbines, December 17, 1986
- MO-7A-2, Emergency Diesel Generator Surveillance, February 21, 2008

1R20 Outage Activities

- Post Event Review Report, January 13, 2008
- GOP-3, Mode 3 > 525 F to mode 2, Revision 25
- GOP-4, Mode 2 to mode 1
- EN #43900, Palisades Plant, January 13, 2008
- EOP 1.0 Standard Post Trip Actions, January 13, 2008
- EOP 2.0 Standard Post Trip Actions, January 13, 2008
- SOP-6, Reactor Control System, Revision 25
- EM-04-24, Palisades Critical Prediction and Critical Approach, Revision 7

1R22 Surveillance Testing

- MO-7A-1, EDG 1-1 Surveillance Test, January 7, 2008
- Condition Evaluation (CE)008810, K6A Service Water Outlet expansion Joint is Being Worn Away by bolt, February 9, 2004
- CR-PLP-2008-00082, Expansion Bolt Contacts Bolt, January 25, 2008
- QI-9, Reactor protective Trip Units, Revision 5
- EN-MA-119, Material Handling Program, Revision 5
- MSM-M-13, Overhead Crane Mechanical Inspection, Revision 30

- FHS-M-23, Movement of Heavy Loads in the Spent Fuel Pool Area, Revision 27
- PAL PWS, Public Warning System Operating Procedure, Revision 18
- NRC Indicator Alert and Notification System reliability, Sept. 2007 through Feb 2008
- QO-14, Inservice test Procedure – Service Water Pumps, March 20 2008
- CR-PLP-2008-01032, leak rate past MV-FW114, FW Heater E-6B has not been Validated Since November 2004, March 3, 2008

1EP6 Emergency Preparedness Drill Observation

- EP Drill and Exercise Performance (DEP), Drill Evaluation Critique, March 19, 2008

2OS1 Access Control to Radiologically Significant Areas

- CR-PLP-2007-04869, An Unposted High Radiation Area Was Discovered While Performing A Follow-Up Survey To Investigate The Electronic Alarming Dosimeter Alarms Documented On CR-2007-04865, October 04, 2007
- CR-PLP-2007-04304, Locked High Radiation Area Lock was Discovered Unlocked and Unguarded on the B Steam Generator Secondary Handholes, September 22, 2007
- CR-PLP-2007-04086, Worker Received Dose Alarm While Performing Fire Watch Activities In The “Old Boronometer Room”, September 18, 2007
- CR-PLP-2007-04638, RP Did Not Generate Condition Reports For All Of The Unanticipated Electronic Dosimeter Dose Rate Alarms That Have Occurred During R-19, September 28, 2007
- CR-PLP-2007-05105, Individual Entered LHRA Without Updating LHRA Briefing, October 10, 2007
- CR-PLP-2007-04340, Locked High Radiation Area Controls Restricting Access To The Regenerative Heat Exchanger Are (via the Primary System Drain Tank platform) May Not Meet Standards For Control, September 23, 2007
- CR-PLP-2007-04762, Two Westinghouse Steam Generator Platform Workers Received An Uptake While Cleaning Primary Manway Stud Holes And Diaphragms On E-50B, October 02, 2007
- CR-PLP-2007-04039, RP and Radworker anomalies were observed on nightshift (September 15, 2007-September 16, 2007), September 17, 2007
- CR-PLP-2007-04149, Westinghouse Employee Performing Reactor Head O-Ring Re-Installation Required Removal Of The Portable Air Purifying Respirator (PAPR), September 19, 2007
- CR-PLP-2007-4002, Higher Than Expected Airborne Radioactivity Concentrations For An Extended Period During 1R19, September 30, 2007
- CR-PLP-2007-04051, A High Radiation Area Posting And Barricade Was Moved On 590’ Containment, September 17, 2007
- CR-PLP-2007-04064, Quality Assurance Identified Eight Locked High Radiations Without Secondary Postings, September 17, 2007
- CR-PLP-2007-04197, Quality Assurance Identified Eight RP And/Or Radworker Deficiencies On Nightshift (September 19, 2007 – September 20, 2007),
- CR-2007-04236, A High Radiation Area Boundary/Posting On 590’ Containment Was Moved By A Radworker, September 20, 2007
- CR-PLP-2007-04245, A Radiation Protection Technician Identified Worker Attempting To Alter Boundary Located 590’ Containment, September 21, 2007
- CR-PLP-2007-05196, Worker Knocked Over A RP Posting, October 13, 2007
- CR-PLP-2007-05198, Radiation Protection Technician Observed High Radiation Area Swing Gate Was Not In The Correct Position, October 13, 2007

- CR-PLP-2007-05595, Worker Received A Dose Rate Alarm Of 140 Mr/Hour While In The East Engineering Safeguards Area, November 04, 2007
- CR-PLP-2008-00229, Incorrect PAPR Chargers Remain In Service After Closure Of CR-PLP-2007-04149, January 17, 2008
- Procedure No HP 7.4, Cleaning, Storing and Maintenance of Respirators, Revision 9
- Procedure No 1.16, Respiratory Protection Program, Revision 2
- PL-RPR-556-206O, On the Job Training and Task Performance Evaluation Guide, Assist Worker in the Use of Powered Air Purifying Respirators, Revision 1
- Draft Report for Evaluation of Internal Dose Assessments by Palisades Nuclear Plant of Workers Exposed in September 2007 Airborne Contamination Event, K.A.L., Inc., January 05, 2008
- Procedure No. HP 2.5, High Radiation Area Entry and Control, Revision 25

2OS2 ALARA Planning and Controls

- CR-PLP-2007-04549, Palisades Cycle 19 Fuel Failures Report, November 20, 2007
- FP-RP-RWP-01, Radiation Work Permit, Revision 7
- RWP 754, Refuel Project – Reactor Vessel Disassembly
- RWP 781, GSI-191 Project – Replace CV-3001 and CV-3002 Replacement
- RWP 765, Chemistry and Radiation Protection
- RWP 823, Valve Repair Activities in Containment
- RWP 756, Refuel Project – Reactor Cavity Decontamination

2PS3 Radiological Environmental Monitoring program and Radioactive Material Control Program

- CR-PLP-2007-04338, Radioactive Contamination Identified On A Shoe That Was Previously Released From The Station, September 22, 2007
- CR-PLP-2007-04381, Radioactive Contamination Identified On A Shoe That Was Previously Released From The Station, September 23, 2007
- 7.15, Contamination Control, Revision 13
- WI-RSD-H-010, Release of Items, Revision 11

4OA1 Performance Indicator Verification

- Esoms, Narrative log search for Reactor trip and Reactor startup, calendar year 2007
- HPSI MSPI report, Palisades First – Fourth Quarter 2007
- Esoms Logs for HPSI and HPSI pumps, 2007
- QO-19, Inservice test Procedure – HPSI Pumps and ESS Check valve operability test, Revision 27
- QO-1, Safety Injection System, Revision 56
- Palisades Nuclear Plant, MSPI Basis Document, June 28, 2007

4OA2 Problem Identification and Resolution

- CR-PLP-2008-01058, Lubricator ML-3037 for VOP-3037 Showing Evidence of Leakage, March 5, 2008
- CR-PLP-2008-01324, Potentially Non-Conservative TS, March 24, 2008
- CR-PLP-2007-06351, CV-3047 Not Closed, December 18, 2008
- Apparent Cause Evaluation for CR-PLP-2007-06351, Revision 0
- CR-PLP-2007-5375, Control Problems with CV-0703, October 20, 2007

- Apparent Cause Evaluation for CR-PLP-2007-5375, Revision 0
- Apparent Cause Evaluation for CR-PLP-2007-0873, Revision 0
- CR-PLP-2007-00873, P-8C Flow Control to E-5oA Bypass failed Open, February 27, 2007
- CR-PLP-2008-1054, MO-3199 has Shown Indications of Internal Guide Wear, March 4, 2008
- Vendor Drawing D-54878, 14”-2216-EMO-SP gate valve, Revision 2
- MOV Performance Data MO 3189, MO3190, MO3198, MO3199, 2007.
- MO 3199 Diagnostic test data EM-28-07, MOV Diagnostic Test Engineering Acceptance, December 21, 2007
- GL 95-07, Pressure Locking and Thermal Binding of Safety Related Power-Operated Gate Valves, August 15, 1995
- CR-PLP-2007-06425, MO3199 tested SAT, Analysis Shows Internal Guide to Disc wear, December 20, 2007
- EA-PLTB-00, Pressure Locking and Thermal Binding Review for Power Operated Gate Valves in Response to GL 95-07, Revision 4
- EA-GL-8910-PPM-01, Analysis Input for GL89-10 Program, Revision 4

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

- LER 05000255/2007-010-00, TSs Action Requirements Not Met for Battery Cell Parameter Outside Allowable Limits, Revision. 0
- ACE evaluation for CR-PLP-2007-6496, Revision 0
- WO00330130, ED02 Replace Cell #43, December 28, 2007
- QE-35, ED01 and ED02 Battery Checks – Quarterly, performed December 27, 2007
- Administrative Procedure 9.20, TS Surveillance and Special test program, Revision 23
- Post Event Review Report, January 13, 2008
- IEEE Standard 484-1996, IEE Recommended Practice for Installation of Vented Lead-Acid Batteries for Stationary Applications
- LER 05000255/2007009-00, Automatic Valve in Emergency Core Cooling System Inoperable in Excess of TS Requirements, Revision. 0
- CR-PLP-2007-05960, CV-3047 Exceeded Stroke Time, November 26, 2007
- CR-PLP-2007-06351, CV-3047 Partially Open, December 18, 2007
- CR-PLP-2008-0151, P-1B Main Feedwater Pump Trip, January 14, 2008
- FP-WM-PLA-01, Work Order Planning Process, Revision 1
- WI-FWS-M-12, Main Feedwater Turbine Maintenance Instruction, Revision 2
- LER 05000255/2008-01-00, Reactor Protection System and Auxiliary Feedwater System Actuation, Revision 0
- FP-WM-PLA-01, Work Order Planning Process, Revision 1

LIST OF ACRONYMS USED

ADAMS	Agency Wide Document and Management System
AFW	Auxiliary Feed Water
ALARA	As Low As Reasonably Achievable
AOV	Air Operated Valve
CFR	Code of Federal Regulations
CR	Condition Report
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FIN	Finding
GOP	General Operating Procedure
HPSI	High Pressure Safety Injection
JPM	Job Performance Measure
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LORT	Licensed Operator Requalification Training
MOV	Motor Operated Valve
MSPI	Mitigating System Performance Indicator
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
PAPR	Powered Air Purifying Respirator
RCA	Radiologically Controlled Area
RWP	Radiation Work Permit
SAT	Systems Approach to Training
SDP	Significance Determination Process
SSC	Systems, Structures, and Components
TS	Technical Specification
URI	Unresolved Item