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

February 7, 2013

Mr. Vito Kaminskas
Site Vice President, Nuclear
FirstEnergy Nuclear Operating Company
Perry Nuclear Power Plant
P. O. Box 97, 10 Center Road, A-PY-A290
Perry, OH 44081-0097

SUBJECT: PERRY NUCLEAR POWER PLANT – NRC INTEGRATED INSPECTION REPORT 05000440/2012005 AND 07200069/2012002

Dear Mr. Kamiskas:

On December 31, 2012, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection and an inspection of the initial operation of the Independent Spent Fuel Storage Installation (ISFSI) at the Perry Nuclear Power Plant Unit 1. The enclosed inspection report documents the inspection results which were discussed on January 10, 2013, with you and other members of your staff.

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

Three NRC-identified and two self-revealed findings of very low safety significance (Green) were identified during this inspection.

All of these findings were determined to involve violations of NRC requirements. Additionally, the NRC determined that one traditional enforcement Severity Level IV violation occurred. This traditional enforcement violation was not identified with an associated finding. The NRC is treating these violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the Enforcement Policy.

If you contest the violations or the significance of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies 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 Perry Nuclear Power Plant.

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

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

Sincerely,

/RA/

Michael A. Kunowski, Chief
Branch 5
Division of Reactor Projects

Docket No. 05000440 and 07200069
License No. NPF-58

Enclosure: Inspection Report 05000440/2012005 and 07200069/2012002
w/Attachment: Supplemental Information

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440 and 72-069
License No:	NPF-58
Report No:	05000440/2012005 and 07200069/2012002
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, Ohio
Dates:	October 1, 2012 through December 31, 2012
Inspectors:	M. Marshfield, Senior Resident Inspector J. Nance, Resident Inspector J. Beavers, Emergency Preparedness Inspector M. Bielby, Senior Operations Engineer J. Corujo-Sandín, Reactor Engineer R. Edwards, Reactor Inspector, Materials Control, ISFSI, and Decommissioning Branch (MCID), Division of Nuclear Materials Safety (DNMS) R. Jickling, Senior Emergency Preparedness Inspector J. Laughlin, Emergency Preparedness Inspector M. Learn, Reactor Engineer, MCID, DNMS C. Moore, Operations Engineer M. Phalen, Senior Health Physicist J. Tapp, Health Physicist, MCID, DNMS
Observer:	E. Denison, Ohio Dept of Health, Bureau of Radiation Protection
Approved by:	Michael A. Kunowski, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000440/2012005; 0720069/2012002; 10/01/2012 – 12/31/2012; Perry Nuclear Power Plant; Maintenance of Emergency Preparedness; Radiological Hazard Assessment and Exposure Controls; Other Activities

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by headquarter and regional inspectors. Six findings were identified by the inspectors. Five of the findings were considered Green non-cited violations (NCVs) of NRC regulations. One of the findings was evaluated as a traditional enforcement Severity Level IV (SL IV) violation. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" dated June 2, 2011. Cross-cutting aspects are determined using IMC 0310, "Components Within the Cross-Cutting Areas," dated October 28, 2011. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy dated June 7, 2012. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the failure to establish appropriate procedures capable of restoring low-pressure coolant injection (LPCI) mode of residual heat removal (RHR), while in the shutdown cooling (SDC) mode, following a loss-of-coolant accident (LOCA) in Mode 3. Specifically, the licensee failed to prescribe procedures which ensured: (1) LPCI could be restored using only safety-related/seismic structures, systems and components; (2) no unanalyzed water hammer event occurred; (3) the equipment used for venting the system were appropriate; and (4) operator safety was maintained. This finding was entered into the licensee's corrective action program and the licensee instituted compensatory actions to declare RHR trains INOPERABLE while aligned to SDC. Additionally, procedures affected are prohibited from use while the plant is in Mode 3.

  The performance deficiency was determined to be more than minor because, if left uncorrected it could have the potential to lead to a more significant safety concern. Specifically, the inspectors had concerns that procedures, as currently written, would have been unsuccessful in restoring LPCI. The finding screened as having a very low safety significance based on a Phase II Significance Determination Process evaluation. The result was a delta core damage frequency less than 1.0E-6/year. The inspectors determined this finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because the licensee did not implement operating experience through changes to the station's process, procedures, and equipment. Specifically, the licensee's evaluation of Information Notice 2010-11 incorrectly concluded sufficient barriers were in place to prevent the occurrence of steam voiding in the RHR system (P.2(b)). Section 4OA5.1.c(1))

- Green. The inspectors identified a finding of very low safety significance and associated non-cited violation of 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” for the failure to ensure adequate test instrumentation was available and used during the performance of periodic venting. This finding was entered into the licensee’s corrective action program and the licensee will revise the affected procedures to require the use of a timepiece.

  The performance deficiency was determined to be more than minor because it was associated with the Mitigating Systems cornerstone attribute of “Procedure Quality: Maintenance and Testing Procedures.” Specifically, by not using adequate test instrumentation to measure the time gas was vented, the licensee introduced further uncertainty to an already inaccurate method. The finding screened as having very low safety significance because the finding involved a design or qualification deficiency that did not result in a loss of operability. Specifically, review of the licensee’s corrective action program documents for resolution of Generic Letter 2008-01 determined that voids had been identified following system restoration (initial fill and vent) while the system was inoperable, and voids identified when the system was online had been significantly below the calculated acceptance criteria. This finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because the licensee did not thoroughly evaluate relevant external operating experience. Specifically, the licensee’s evaluation of Nuclear Energy Institute 09-10, Revision 0, failed to identify the importance of having adequate venting time information when quantifying vented voids (P.2(a)). Section 4OA5.1.c(2))

Cornerstone: Barrier Integrity

Green. A finding of very low safety significance and associated non-citied violation of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed for the failure to perform adequate maintenance on the single-failure-proof fuel handling building (FHB) crane used to handle dry storage casks containing spent nuclear fuel. The licensee corrected the issue prior to conducting lifts containing spent nuclear fuel and entered it into their corrective action program (Condition Reports 2012-13234, 2012-13315, and 2012-12933).

The inspectors determined the performance deficiency was more than minor in that it affected the Human Performance attribute (maintenance performance) of the Barrier Integrity cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radioactive releases caused by accidents or events. Additionally, if left uncorrected, a malfunction of the FHB crane could lead to a more significant safety concern. Based on answering “No” to all the screening questions in IMC 0609, Appendix A, Exhibit 3, “Barrier Integrity Screening Questions,” the finding was determined to be of very low safety-significance (Green). This finding had a cross-cutting aspect in the area of Human Performance, Resources, because the licensee failed to have complete, accurate, and up-to-date procedures that ensured personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee failed to have maintenance procedures that ensured the FHB crane would be capable of performing its single-failure-proof design functions that assure nuclear safety (H.2(c)). (Section 4OA5.3(1))

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a finding of very low safety significance with an associated non-citied violaiton of 10 CFR 50.54(q)(2) for the failure to follow the Perry Nuclear Power Plant Emergency Plan that uses a standard emergency classification and action level scheme. Specifically, on June 7, 2012, Perry personnel failed to classify an Unusual Event for an unexpected increase in plant radiation levels when health physics surveys indicated an increase by a factor of 1000 times over normally expected area radiation levels. On June 14, 2012, the licensee initiated CR 2012-09729 to determine why an Unusual Event was not classified for the June 3, 2012, resin spill, and why there was a failure to classify the unexpected increase in plant radiation levels identified in surveys of the 574' elevation of the radwaste building on June 7. On November 29, 2012, the licensee initiated CR 2012-18622 to identify and investigate reasons for the Unusual Event requirements.

  The failure to implement the emergency plan and classify an Unusual Event was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it affected the Emergency Response Organization performance attribute of the Emergency Preparedness cornerstone and adversely affected the cornerstone objective to ensure the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Using Inspection Manual Chapter 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Attachment 1, the finding was determined to have very low safety-significance (Green) because the actual event implementation problem was associated with an Unusual Event. This finding had a cross-cutting in the area of Problem Identification and Resolution, Corrective Action Program, for evaluation and extent of condition (P.1c)). Specifically, Perry personnel failed to properly evaluate and classify an Unusual Event for the June 3, 2012, resin spill conditions in CR 2012-09447, dated June 7, 2012, and CR 2012-09729, dated June 14, 2012. (Section 1EP5.1)

Occupational Radiation Safety

- Green. A finding of very low safety significance and associated non-cited violation of 10 CFR 20.1501 was self-revealed for the failure of the licensee to make surveys to ensure compliance with 10 CFR 20.1601 and Technical Specification 5.7.2 from June 3 through June 7, 2012. Specifically, the licensee failed to evaluate the radiological conditions and potential radiological hazards associated with the spill of radioactive resins on the 574' elevation of the radioactive waste processing building that resulted in the failure to properly barricade and conspicuously post the area as required by 10 CFR 20.1601 and Technical Specification 5.7.2. The area was found to be accessible to personnel with radiation levels such that a major portion of the whole body could receive in 1 hour a dose greater than or equal to 1000 millirem. Corrective actions included performing complete radiological surveys of the area, posting and controlling the area as required by licensee Technical Specifications. These actions were completed on June 7, 2012.

  The inspectors determined that this finding was more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety cornerstone and adversely affected the associated cornerstone objective of protecting worker health and safety from exposure to radiation. Specifically, not barricading and conspicuously posting high radiation areas may result in unnecessary and unplanned radiation exposures to workers. The inspectors reviewed the finding in accordance with Inspection Manual Chapter 0609, Appendix C, Occupational Radiation Safety Significance Determination Process, and determined that the finding was of very low safety significance because the finding did not involve as-low-as-is-reasonably-achievable (ALARA) planning or work controls, there was no overexposure or substantial potential for an overexposure, nor was the licensee's ability to assess worker dose compromised. The inspectors concluded that the most significant contributor to the finding was in the cross-cutting area of Human Performance with the component of decision making (H.1.(b)). (Section 2RS1)

Other Findings

Severity Level IV. The inspectors identified a Severity Level IV non-cited violation of very low safety significance of 10 CFR Part 72.150, "Instructions, Procedures, and Drawings," for the failure by the licensee to follow procedures that ensured the safe loading of a dry fuel storage canister into a storage cask. The licensee corrected the issue to restore compliance with the procedure and placed the concern in its corrective action program (CR 2012-15087).

The violation was determined to be of more than minor significance using IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," Example 2f, in that a procedural requirement was not met and the actual distance between the HI-STORM storage cask containing the dry fuel storage canister and the end of the rails on which the cask would be moved was less than the analyzed distance required to ensure safe transport operations. The inspectors determined that the violation could be evaluated using Section 6.5.d.3 of the NRC Enforcement Policy, as a Severity Level IV violation, in that the licensee failed to follow procedures affecting the safe transport of a HI-STORM. Cross-cutting aspects are not assigned to traditional enforcement violations. Since this violation was dispositioned using traditional enforcement, a cross-cutting aspect is not applicable. (Section 4OA5.3(2))

B. Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period at 100 percent power. Plant power was lowered only for rod pattern adjustments and in support of surveillance testing requirements for the remainder of the inspection period and was at 100 percent power at the end of the quarter.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

Winter Seasonal Readiness Preparations

a. Inspection Scope

The inspectors conducted a review of the licensee's preparations for winter conditions to verify that the plant's design features and implementation of procedures were sufficient to protect mitigating systems from the effects of adverse weather. Documentation for selected risk-significant systems was reviewed to ensure that these systems would remain functional when challenged by inclement weather. During the inspection, the inspectors focused on plant specific design features and the licensee's procedures used to mitigate or respond to adverse weather conditions. Additionally, the inspectors reviewed the Updated Safety Analysis Report (USAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Cold weather protection, such as heat tracing and area heaters, were verified to be in operation where applicable. The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into the CAP in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment to this report. The inspectors' reviews focused specifically on the condensate storage tank systems, including heat tracing, and building heating systems, due to their risk significance or susceptibility to cold weather issues.

This inspection constituted one winter seasonal readiness preparations sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- emergency service water (ESW);
- reactor core isolation cooling; and
- Division 1, emergency diesel generator (EDG).

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

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

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On November 29, 2012, the inspectors performed a complete system alignment inspection of the motor control center, switchgear heating, ventilation, air-conditioning and battery room exhaust ventilation system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding work orders was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

a. Inspection Scope

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

- Fire Zones 0FH-1, 0FH- 2A, and 0FH2B (Fuel Handling Building 574' / 585' elevations and Fuel Handling Building 599' elevation);
- Fire Zones DG-1D (Diesel Generator Building 620' 6" / 646' 6" elevations);
- Fire Zone 5A (Control Complex 654' elevation including Unit 1 and Unit 2 Control Rooms);
- Fire Zones SB-604 and SB-620 (Service Building 604' / 620' elevations); and
- Fire Zones 4 and 5 (Intermediate Building 654' and 682' elevations).

The inspectors reviewed areas to assess if the licensee had implemented a fire protection program that adequately controlled combustibles and ignition sources within the plant, effectively maintained fire detection and suppression capability, maintained passive fire protection features in good material condition, and implemented adequate compensatory measures for out-of-service, degraded or inoperable fire protection equipment, systems, or features in accordance with the licensee’s fire plan.
The inspectors selected fire areas based on their overall contribution to internal fire risk as documented in the plant’s Individual Plant Examination of External Events with later additional insights, their potential to impact equipment which could initiate or mitigate a plant transient, or their impact on the plant’s ability to respond to a security event.
Using the documents listed in the Attachment to this report, the inspectors verified that fire hoses and extinguishers were in their designated locations and available for immediate use; that fire detectors and sprinklers were unobstructed; that transient material loading was within the analyzed limits; and fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors also verified that minor issues identified during the inspection were entered into the licensee’s CAP.
Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

a. Inspection Scope

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

- Auxiliary Building elevations 574' and 568'; and
- Control Complex elevations 574' and 599'.

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

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

a. Inspection Scope

The inspectors reviewed the licensee's testing of the emergency closed cooling system 'B' heat exchanger to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. Inspectors also verified that test acceptance criteria considered differences between design conditions, and testing conditions. Documents reviewed for this inspection are listed in the Attachment to this document.

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

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On November 14, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training to verify that operator performance was adequate, evaluators were identifying, and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11-05, and satisfied the inspection program requirement for the resident inspectors to observe a portion of an in-progress annual requalification operating test during a training cycle in which the NRC did not observe it during the biennial portion of this IP.

b. Findings

No findings were identified.

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

a. Inspection Scope

On October 26, the inspectors observed the licensee in the control room conduct a down-power to approximately 65 percent reactor thermal power to perform a rod pattern adjustment. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

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

- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of procedures;
- control board and equipment manipulations;
- oversight and direction from supervisors; and
- ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications.

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

This inspection constituted one quarterly licensed operator heightened activity/risk sample as defined in IP 71111.11-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the overall pass/fail results of the Biennial Written and the Annual Operating Test, administered by the licensee from October 22 through December 7, 2012, required by 10 CFR 55.59(a). The results were compared to the thresholds established in IMC 0609, Appendix I, "Licensed Operator Requalification Human Performance Significance Determination Process," to assess the overall adequacy of the licensee's Licensed Operator Requalification Training (LORT) program to meet the requirements of 10 CFR 55.59.

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

b. Findings

No findings were identified.

.4 Biennial Review (71111.11B)

a. Inspection Scope

The following inspection activities were conducted during the weeks of November 26 and December 3, 2012, to assess: 1) the effectiveness and adequacy of the licensee's implementation and maintenance of its systems approach to training (SAT) based LORT program, put into effect to satisfy the requirements of 10 CFR 55.59; 2) conformance with the requirements of 10 CFR 55.46 for use of a plant referenced simulator to conduct operator licensing examinations and for satisfying experience requirements; and 3) conformance with the operator license conditions specified in 10 CFR 55.53. The documents reviewed are listed in the Attachment to this report.

- <u>Problem Identification and Resolution (10 CFR 55.59(c); SAT Element 5 as defined in 10 CFR 55.4)</u>: The inspectors evaluated the licensee's ability to assess the effectiveness of its LORT program and its ability to implement appropriate corrective actions to maintain its LORT program up-to-date. The inspectors reviewed documents related to the plant's operating history and associated responses (e.g., plant issue matrix and performance review reports; recent examination and inspection reports (IRs); and licensee event reports (LERs)). The inspectors reviewed the use of feedback from operators, instructors, and supervisors as well as 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.

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

  - The inspectors reviewed the methodology used to construct the examination including content, level of difficulty, and general quality of the examination/test materials. The inspectors also assessed the level of examination material duplication from week-to-week for both the operating tests and written examinations administered in 2012. The inspectors reviewed a sample of the written examinations and associated answer keys to check for consistency and accuracy.

  - The inspectors observed the administration of the annual operating test and biennial written examination to assess the licensee's effectiveness in conducting the examinations, including the conduct of pre-examination briefings, evaluations of individual operator and crew performance, and post-examination analysis. The inspectors evaluated the performance of one crew in parallel with the facility evaluators during three dynamic simulator scenarios, and evaluated various licensed crew members concurrently with facility evaluators during the administration of several Job Performance Measures (JPMs).

  - The inspectors assessed the adequacy and effectiveness of the remedial training conducted since the last requalification examinations and the training planned for 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.

- <u>Conformance with Examination Security Requirements (10 CFR 55.49)</u>: The inspectors conducted an assessment of the licensee's processes related to examination physical security and integrity (e.g., predictability and bias) to verify compliance with 10 CFR 55.49, "Integrity of Examinations and Tests." The inspectors reviewed the licensee's examination security procedure, and observed the implementation of physical security controls (e.g., access restrictions and simulator Input/Output controls) and integrity measures (e.g., security agreements, sampling criteria, bank use, and test item repetition) throughout the inspection period.

- Conformance with Simulator Requirements (10 CFR 55.46): The inspectors assessed the adequacy of the licensee's simulation facility (simulator) for use in operator licensing examinations and for satisfying experience requirements. The inspectors reviewed a sample of simulator performance test records (e.g., transient tests, malfunction tests, scenario based tests, post-event 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 corrective action process to ensure that simulator fidelity was being 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.

- Conformance with Operator License Conditions (10 CFR 55.53): The inspectors reviewed the 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. Additionally, medical records for six licensed operators were reviewed for compliance with 10 CFR 55.53(i).

This inspection constitutes one biennial licensed operator requalification inspection sample as defined in IP 71111.11-05.

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- reactor recirculation system;
- reactor protection system Division 'A'; and
- diesel generator starting air system.

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

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;

- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for structures, systems, and components (SSCs)/functions classified as (a)(2), or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

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

This inspection constituted three quarterly maintenance effectiveness samples as defined in IP 71111.12-05.

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- dry cask storage locked high radiation area controls;
- emergency core cooling system Division 1 outage; and
- reactor core isolation cooling steam supply first drain shutoff valve steam leak.

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

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

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- torquing of cam shaft cover bolts on Division 2 EDG;
- personnel airlock, elevation 603' supply air outboard isolation valve failed stroke time surveillance test;
- flow control valve 'A' operability during hydraulic power unit 'A-2' maintenance activities; and
- ESW pump house ventilation fan '1B' operability determination.

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

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

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the following modifications:

- essential service water pump 'A' "hot short" breaker; and
- online Noble Chemistry application.

The inspectors reviewed the configuration changes and associated 10 CFR 50.59 safety evaluation screening against the design basis, the USAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected systems. The inspectors, as applicable, observed ongoing and completed work activities to ensure that the modifications were installed as directed and consistent with the design control documents; the modifications operated as expected; post-modification testing adequately demonstrated continued system operability, availability, and reliability; and that operation of the modifications did not impact the operability of any interfacing systems. As applicable, the inspectors verified that relevant procedure, design, and licensing documents were properly updated. Lastly, the inspectors discussed the plant modification with operations, engineering, and training personnel to ensure that the individuals were aware of how operation with the plant modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

This inspection constituted one temporary modification sample and one permanent plant modification samples as defined in IP 71111.18-05.

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- replacement of the RHR 'A' heat exchanger bypass valve control switch;
- Division 1 EDG hot short modification on EH-11;
- ESW 'B' breaker pocket ground truck installation and removal;
- RHR 'B' pump electrical maintenace retest; and
- Division 2 EDG risk-informed allowed outage time annual maintenance activities.

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 TSs, the USAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with post-maintenance tests to determine whether the licensee was identifying problems and entering them in the CAP and that the problems were being corrected commensurate with their importance to safety. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

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

- Surveillance Instruction (SVI)-C41-T2001A; Standby Liquid Control Pump 'A' Pump and Valve Operability Testing (Inservice Testsing (IST));
- SVI-E51-T2001; Reactor Core Isolation Cooloing Pump and Valve Operability Testing (IST);
- SVI-M15-T1239B; Annulus Exhaust Gas Treatment Train 'B' Operability Test (routine); and
- Daily unidentified leakage TS surveillance and operator assessment process (Reactor Coolant System (RCS) leakage).

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

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

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

Documents reviewed are listed in the Attachment to this report.

This inspection constituted one routine surveillance testing sample, two inservice testing samples, and one RCS leak detection sample as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings were identified.

1EP4 Emergency Action Level and Emergency Plan Changes (IP 71114.04)

a. Inspection Scope

The NRC's Nuclear Safety and Incident Response (NSIR) headquarters' staff performed an in-office review of the latest revisions of the Emergency Plan and various Emergency Plan Implementing Procedures located under ADAMS accession numbers ML12096A030, ML12096A031, ML12138A279, ML12152A268, ML12307A428, and ML12307A436, as listed in the Attachment.

The licensee transmitted the Emergency Plan Implementing Procedure revisions to the NRC pursuant to the requirements of 10 CFR Part 50, Appendix E, Section V, "Implementing Procedures." The NRC review was not documented in a safety evaluation report and did not constitute approval of licensee-generated changes; therefore, this revision is subject to future inspection. The specific documents reviewed during this inspection are listed in the Attachment.

This Emergency Action Level (EAL) and Emergency Plan changes inspection constituted one sample as defined in IP 71114.04-05.

b. Findings

No findings were identified.

1EP5 Maintenance of Emergency Preparedness (71114.05)

.1 Maintenance of Emergency Preparedness

a. Inspection Scope

The inspectors performed in-office and on-site reviews of site procedures, documents, and corrective actions related to the June 3, 2012, resin spill event and limited apparent cause evaluation to determine compliance with 10 CFR 50.54(q)(2). Processes describing identification and classification were discussed with emergency preparedness, fleet management, site management, and site emergency response organization personnel. Documents reviewed are listed in the Attachment to this report.

This maintenance of emergency preparedness inspection constituted zero samples as defined in IP 71114.05-06.

b. Findings

Introduction: The inspectors identified a finding of very low safety significance with an associated NCV of 10 CFR 50.54(q)(2) for the failure to follow the Perry Nuclear Power Plant (PNPP) Emergency Plan that uses a standard emergency classification and action level scheme. Specifically, on June 7, 2012, PNPP personnel failed to classify an Unusual Event for an unexpected increase in plant radiation levels when surveys indicated an increase by a factor of 1000 times over normally expected area radiation levels. On June 14, 2012, the licensee initiated CR 2012-09729, a limited apparent cause evaluation, to determine why an Unusual Event was not classified for the June 3, 2012, resin spill, and to determine why the unexpected increase in plant radiation levels identified in surveys on June 7 of the 574' elevation of the radwaste (RW) building was not classified.

Description: On June 3, 2012, PNPP personnel conducted a transfer of spent resins and encountered problems which resulted in a resin spill on the 574’ elevation of the RW building. No radiation surveys were conducted at this time.

On June 7, a radiation survey of the 574' elevation of the RW building area was conducted and documented in CR 2012-09447. The CR problem statement indicated that dose rates were discovered greater than 1000 millirem per hour at 30 centimeters in the 574' elevation of the RW building general access hallway. Actions taken by radiation protection (RP) after the survey included posting the area doors with appropriate signs and barriers, and notification of management. Senior Reactor Operator (SRO) reviews and comments in CR 2012-09447 concluded no Emergency Plan (Emergency Action Level) entry criteria had been met due to survey “general area access” dose rates being 650 millirem per hour and did not show an increase by a factor of 1000 times over the highest general area dose rate of 1.2 millirem per hour.

On June 14, CR 2012-09729 was initiated as a limited apparent cause evaluation for an Unusual Event not declared for the 574’ elevation resin spill. The CR problem statement indicated that the increase in dose rates associated with the spill, documented in CR 2012-09477, met the Emergency Plan entry criteria for EAL GU1 and no Unusual Event was declared. Details identified that dose rate surveys taken prior to the spill in the resin spill area of the corridor were 0.3, 0.6, and 0.2 millirem per hour at 30 centimeters. The post-resin spill dose rate surveys in the corridor were 600, 1200, and 1500 millirem per hour at 30 centimeters, respectively.

Fleet Oversight determined the GU1 entry criteria were met when Health Physics surveys indicated an increase by a factor of 1000 times normally expected area radiation levels and that the increase was not due to the start-up and operation of plant equipment or systems within design parameters, planned movement of radioactive materials, or the planned movement of shielding.

Condition Report 2012-09729 concluded, in an independent review of the information, that the entry criteria for EAL GU1, Unexpected Increased Plant Radiation Levels, had not been met. The SRO conclusion was based on highest general area dose rates, which were 1.2 millirem per hour pre-spill and 650 millirem per hour post-spill. The SRO review did not evaluate the 30-centimeter area radiation level in the pre- and post-surveys.

The PNPP Emergency Plan, Unusual Event initiating condition for EAL GU1, Unexpected Increase in Plant Radiation Levels, entry criteria states, in part, surveys indicate an increase by a factor of 1000 times over normally expected area radiation levels and the radiation level increase cannot be attributed to the start-up and operation of plant equipment or systems within design parameters, the planned movement of radioactive materials, or the planned movement of shielding. The technical basis in Preparedness Support Instruction (PSI)-0019 for EAL GU1 states, in part, unplanned increases in in-plant radiation levels represent a degradation in the control of radioactive material and represent a potential degradation in the level of safety of the plant.

The NRC inspectors identified that EAL GU1 was not restricted by defined terms including "general area" dose rates, "waist-level" dose rates, or "general area access" dose rates. Using available documents and interviews, the inspectors concluded that the plant personnel failed to implement the PNPP Emergency Plan EAL GU1 to classify the resin spill event when presented with survey results indicating an increase by a factor of 1000 times over normally expected area radiation levels. This event was an unexpected spill that was an unplanned increase in plant radiation levels, represented a degradation in the control of radioactive material, and represented a potential degradation in the level of safety of the plant.

This issue was entered into the licensee's CAP as CR 2012-18622, dated November 29, 2012.

<u>Analysis</u>: The failure to implement the emergency plan and declare an Unusual Event was a performance deficiency. The performance deficiency was more than minor, and therefore a finding, because it was associated with the Emergency Response Organization performance attribute of the Emergency Preparedness cornerstone and adversely affected the cornerstone objective to ensure the licensee is capable of implementing adequate measures to protect the health and safety of the public in the event of a radiological emergency. Using IMC 0609, Appendix B, "Emergency Preparedness Significance Determination Process," Attachment 1, dated February 24, 2012, the finding was determined to have very low safety significance (Green) because the actual event implementation problem was associated with an Unusual Event. The primary cause of this finding was related to the cross-cutting aspect of evaluation and extent of condition under the Problem Identification and Resolution cross-cutting area, Corrective Action Program component. Specifically, Perry personnel failed to properly evaluate the June 3, 2012, resin spill conditions in CR 2012-09447, dated June 7, 2012, and CR 2012-09729, dated June 14, 2012 (P.1.(c)).

<u>Enforcement</u>: Title 10 CFR Part 50.54(q)(2) states, in part, that a licensee shall follow an emergency plan that meets the requirements in Appendix E to this part and the planning standards of § 50.47(b). Perry Nuclear Power Plant Emergency Plan, Revision 35, states, in part, the emergency plan will be put into effect whenever a potentially hazardous situation or radiological emergency is identified. The values

shown in Table 4-1 (EALs) are those which an event must be classified in accordance with the guidance set forth. Emergency Action Level GU1 requires an Unusual Event be declared for surveys indicating an increase by a factor of 1000 times over normally expected area radiation levels, and in-plant radiation level increase cannot be attributed to the start-up and operation of plant equipment or systems within design parameters, the planned movement of radioactive materials, or the planned movement of radioactive shielding.

Contrary to the above, the licensee personnel failed to classify an Unusual Event after the GU1 EAL threshold had been reached. Specifically, on June 7, 2012, PNPP personnel failed to classify an Unusual Event for an unexpected increase in plant radiation levels when surveys indicated an increase by a factor of 1000 times over normally expected area radiation levels. The licensee initiated CR 2012-09729, a limited apparent cause evaluation, on June 14, 2012, to determine why an Unusual Event was not declared, and to determine why the unexpected increase in plant radiation levels identified in surveys on June 7. On November 29, 2012, the licensee initiated CR 2012-18622, to identify and investigate reasons for the Unusual Event requirements. Because this finding is of very low safety significance and has been entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000440/2012005-01; Failure to Classify an Unusual Event).**

This maintenance of emergency preparedness inspection constituted zero samples as defined in IP 71114.05-06.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

The inspection activities supplement those documented in NRC IR 05000440/2012003 and constitute one complete sample as defined in IP 71124.01-05.

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

b. Findings

Introduction: The inspectors reviewed one self-revealed finding of very low safety significance and an associated NCV of 10 CFR 20.1501 for the failure to perform surveys to ensure compliance with 10 CFR 20.1601 and TS 5.7.2 from June 3 through June 7, 2012. The licensee failed to evaluate the radiological conditions and potential radiological hazards associated with the spill of radioactive resins on the 574' elevation of the RW processing building (RW 574') that resulted in the failure to properly barricade and conspicuously post the area as required by 10 CFR 20.1601 and TS 5.7.2. The area was found to be accessible to personnel with radiation levels such that a major portion of the whole body could receive in 1 hour a dose greater than or equal to 1000 mRem.

Description: Degraded material condition of the floor drain system had existed for an extended period of time in the RW building. Although a non-safety related system, the station had documented since 2007 chronic RW equipment processing issues, including leaks on the floor drain flatbed filter resin housing, leaking pumps, leaking pump seals, leaking valves, and blocked and/or backed up floor drains. The inspectors also reviewed various licensee documents that were specific to resin entering the floor drain system sumps dating back to 2007. Because of unavailability and maintenance issues associated with various RW processing equipment, the condensate backwash settling tanks (CBSTs) were used to process multiple resin streams, including condensate and spent fuel pool (SFP) clean up resins. This resulted in the varying of radiological dose rates and varying radiological conditions, at any given time, in plant areas associated with CBST system equipment.

The RW 574' was initially properly posted with selected open areas of the floor controlled as a high radiation area with administratively secured swing gates maintaining access to the area. The recent material conditions issues associated with resin spills on RW 574' began on May 29, 2012. On this date, both floor drain system sump pumps tripped on an overload signal and would not operate. A non-licensed operator monitored the area for flooding during the night of May 29 and observed water coming out of the floor drains. Approximately 1 to 2 inches of water was noted on RW 574' floor by operations staff. Maintenance personnel then performed troubleshooting on the floor drain system sump pumps and identified binding due to resin buildup in the sump. A temporary sump pump was then installed in the floor drain system sump with its discharge routed through the RW 574' east open hallway and into the RW equipment drain system sump. The RP staff installed an area radiation monitor (AMP-100) to monitor radiation levels on the temporary hose between the RW floor drain system sump to equipment drain system sump. Ambient dose rates in the area as indicated by the AMP-100 were nominally 0.2 mRem/hour. Operation of the temporary sump pump required a non-licensed operator to manually start and stop the pump by plugging in the power supply.

Material condition issues on RW 574' continued to degrade into the early morning of June 2, 2012. The operations department identified that resin in the sump may have also been clogging the temporary sump pump. The RW floor drain system sump high-level alarm was locked in, and the temporary sump pump was running continuously in order to keep up with floor drain system (water and resin) inputs. The licensee noted issues with manual operation of the temporary sump pump at about 0055 hours when approximately 1 to 3 inches of water was found on the RW 574' floor with water overflowing into the building stairwell and elevator shaft. The RP staff was aware of the general degraded operational conditions on RW 574'. An RP supervisor expressed concerns to operations management about the water that was observed flowing into the RW stairway and elevator shaft. The RP supervisor notified operations of the need to stop processing radioactive waste. However, no investigation of the source of the water leakage occurred. As a result of the flooding, the RP staff expanded the RW 574' highly contaminated area by extending the boundary postings to inside the east and west stairwells. Once radioactive waste processing was stopped, the elevated high water level condition on RW 574' abated. A main control room operations supervisor instructed the RW operations supervisor to inform him of any high level alarms in the floor drain system so that a non-licensed operator could be sent to operate the temporary sump pump.

The inspectors identified three examples of the licensee's failure to perform radiological surveys and evaluate the potential radiological hazards in response to known degradation of radiological and material conditions on RW 574'. This resulted in an ongoing non-compliance to the station's high radiation area access and control program. Each example represents information or conditions that if reasonably acted upon would have ended the on-going noncompliance to station technical specifications. Specifically:

1. <u>Loss of Condensate Backwash Settling Tank Inventory</u>:

   On June 2, 2012, at 2244 hours the area radiation monitors on RW 574' increased from a nominal 0.3 millirem/hour to 0.5 millirem/hour. No area radiation monitors went into alert or alarm status. These radiation monitors were not in the immediate area of the resin spill. Consequently, small increases in monitor readings were potentially indicative of more significant increases in open area dose rates in the 574' east-west hallway. The licensee determined that this was most likely the time when material conditions degraded in the east-west corridor hallway of RW 574' such that area dose rates elevated to meet the conditions requiring TS locked high radiation area controls.

   On June 3, at about 0025 hours, an RP technician was sent to RW 574' to obtain a routine air sample for monitoring for airborne radioactivity. The RP technician noted water and resin on the floor, including resin buildup beyond what had become the normal floor drain backup in the east-west corridor hallway. The RP technician immediately exited the area and notified the shift RP technician and the RW operations supervisor.

   At approximately 0337 hours, an RP technician and a non-licensed operator entered RW 574' to monitor the operation of the temporary sump pump. Prior to entering the area, the RP technician noted that the area radiation monitor (AMP-100) meter was reading 40 millirem/hour. According to the technician, this reading was significantly higher than expected. The RP technician and non-licensed operator observed water and resin in the area near the east stairwell and exited the area. No additional radiation area surveys were performed, and conditions in the RW 574' east-west corridor hallway were not observed.

   At approximately 0400 hours, the RW operations supervisor observed a larger than expected level decrease in CBST tank inventory. The RW operations supervisor called the RP control point and informed the RP technicians that he believed there was a failed seal on the CBST transfer pump. The suspected seal failure was based on the operator's previous experience with degraded RW processing equipment. An RP technician restricted access to RW 574' and informed the on-duty RP supervisor.

   The RW operations supervisor took action to secure the CBST tank recycling process. At 0415 hours the process was secured and the level decrease in the CBST stopped. At approximately 0500 hours, the RP supervisor briefed the outage control center (OCC) of the flooding conditions in RW 574' during the routine shift briefing. [The OCC was staffed with senior plant managers in support of a plant down power and reactor water clean-up system outage.] In followup interviews it was determined that the OCC Shift Outage Director assumed the condition in RW to be within station norms and the result of the chronic issue of the backup of the RW floor drain system.

At approximately 0600 hours, the RP turnover from night-shift to day-shift took place. However, the night-shift RP Supervisor and RP technicians provided only general information to the on-coming RP staff about the flooding and resin on RW 574' floor. There was no mention of the changed operational, material, or radiological conditions, or the extent of the resin spill.

Also, on June 3, at approximately 0923 hours, a non-licensed operator entered RW 574' and stopped when he saw water/resin on the floor. As a direct result of the licensee's lack of radiological response to the degraded material conditions in the area, the non-licensed operator had unencumbered access to the elevated dose rate areas in the east-west hallway corridor area without being made aware of the radiological hazards in this area. This represents a concurrent failure to comply with the licensee's TS for locked high radiation area access controls.

2. <u>Increased Radiation Levels and Degraded Material Conditions Identified</u>:

   On day-shift June 3, 2012, different RP technicians and RP supervisors were on duty. Similar to the observations identified in NRC IR 05000440/2012009, this group of RP technicians and RP supervisors did not receive an adequate turnover regarding the material and radiological condition of RW 574' from their respective night-shift counterparts. Consequently, any information that the day-shift RP staff received or determined during their on-shift activities was compared to their understanding of what had become the usual and customary status for RW 574'. In terms of assessing licensee performance, information identified and reviewed by the day-shift RP staff was separate and discrete from the previous night-shift activities.

   At approximately 1442 hours, an RP technician and a maintenance services worker entered the floor drain system sump room to investigate why the temporary sump pump was not working and to install a new temporary sump pump. The individuals entered RW 574' through the east stairwell and attempted to remove the temporary sump pump, which would not move. The workers could see resin down in the sump. They then attached a rope to the pump and proceeded to pull it out. When the pump was removed, it was surveyed and placed in a bag. A new pump was then installed and started, but it was not pumping water. After two failed attempts, the pump was pulled out of the sump and found to be covered in resin. The resin was approximately 2 feet from the top of the sump. A new hose was installed and routed out of the room and into the equipment drain sump. At this time, there was 2 inches of standing water on RW 574'. The temporary sump pump was eventually started. The individuals stayed in the area to monitor the pumping process for about 5 minutes before exiting. While waiting, the RP technician looked down the east-west corridor hallway and observed resin outside the CBST room. The RP technician also noted that he was in an 18 millirem/hour field while standing outside the RW equipment drain system room. General area dose rates in this area were expected to be nominally less than 1 millirem/hour.

   Followup interviews with the RP technician indicated that he did not enter the east-west corridor hallway to perform a survey because he believed that he did not have a radiation work permit that would allow him to perform the task. Consequently, after leaving the area, the RP technician reported the unexpected material conditions to the on-duty RP supervisor. No RP personnel went down to RW 574' to radiologically assess the area, and no actions were taken to restrict access to the area of concern. Additionally, the anomalous 18 millirem/hour dose rate conditions outside the RW equipment drain system room were not documented on a radiological survey.

The next day, on June 4, at approximately 0600 hours, the RP technician who was the person that observed the resin in the east-west hallway the day before, interrupted the RP supervisor's turnover meeting and expressed his concern with the resin build-up and degraded radiological conditions on RW 574'. The RP technician was acknowledged, informed that RP department supervision was aware of the issue. No additional surveys were performed at this time.

Later that morning at approximately 0936 hours, another RP technician entered RW 574' to perform a pre-job survey for operations to perform valve manipulations. The RP technician documented anomalous dose rates of 8 millirem/hour general area dose rates near the temporary floor drain system sump hose in the east access area. An RP decontamination supervisor was present with the RP technician to assess the extent of the anticipated resin cleanup. The RP decontamination supervisor realized that the scope of the resin spill was more extensive than initially believed. The RP decontamination supervisor took pictures of the area. After leaving the area, the RP decontamination supervisor notified the RP department manager of the extent of the spill. No radiological surveys were performed as a result of the anomalous radiological survey results or because of the now recognized expansiveness of the resin spill.

That evening at approximately 1845 hours, another RP technician entered the area with a non-licensed operator to perform valve manipulations in the RW 574' east access area. This RP technician observed a red colored glaze in the east-west hallway corridor, but no entry was made into the hallway and again no survey was performed.

3. <u>Organizational Under-Response to Anomalous Conditions</u>:

Perry TSs are specific to the roles and responsibilities of the plant manager and the qualifications and responsibilities of the RP manager position on site. The plant manager is responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant. The material condition of the RW building and RW processing systems had been degraded for several years. Although RW processing equipment is non safety-related, the degraded performance of this equipment had consequences. In this case, the consequence was the radiological impact of the failed equipment on plant operations. The degraded material conditions necessitated plant staff to incorporate "workarounds" into their daily operation of the RW facility. These "workarounds" in RW were not being tracked on the operations department "workaround list." Consequently, resolution of the RW "workarounds" was a relatively low station priority. The degraded material condition of the RW processing equipment was allowed by senior plant management to continue, and this essentially desensitized plant staff to the anomalous RW equipment performance issues. The inspectors determined through interviews with plant staff that backups of the RW floor drain system had become an accepted part of RW processing.

The RP manager is responsible for overall station radiation safety, and as a license condition, must have sufficient organizational freedom to ensure their independence from plant operating pressures. The significance that the NRC places on the roles and responsibilities of the plant manager and the RP manager at nuclear power plants is readily apparent, given that these position requirements, roles, and responsibilities were explicitly embedded as conditions in the plant's operating license.

The inspectors observed that there was a 5-day delay in assuring that a complete radiological characterization was performed, and assuring that timely radiological controls were in place on RW 574'. The inspectors determined that the operations shift manager, RW shift supervisor, and the RP manager were all generally aware of radiological and degraded material conditions on RW 574' as early as June 2, 2012. The OCC shift outage director was made aware of the degraded radiological and material conditions on RW 574' on June 3. The inspectors concluded that the plant management staff initially assessed the radiological implications of the RW 574' resin spill and floor drain system backups as consistent with previous spills in the area. The management staff assumed that the current material and radiological condition issues were another backup of the floor drain system and failed to verify actual in-field conditions. The organizational focus was on radiological contamination and airborne controls for the resin spill and floor drain system backup. The RP manager assumed that the radiological postings and controls that were in place prior to the resin spill were sufficient to account for these changed conditions; however, he did not validate his assumptions.

On June 4, an RP decontamination supervisor specifically met with the RP manager and showed him pictures of the resin spill area in order to communicate the expansiveness of the spill. The RP decontamination supervisor also expressed concern with the radiological contamination in the area. No discussion took place on the potential for elevated dose rates in the area. The RP manager assumed that a complete radiological survey of the area had been performed. No action was taken at this time to assure that a comprehensive radiological assessment of the area existed. The RP manager directed the generation of an area recovery plan with a job-specific radiation work permit that included controls for potential airborne contamination.

At the next morning's RP department shift briefing (Tuesday, June 5, 2012, at 0630 hours), the RP technicians continued to express concerns about the degraded radiological and material conditions on RW 574'. The RP manager was present at this briefing. It was recognized that radiological conditions in the area were uncharacterized and the RP manager gave verbal instruction that no personnel enter RW 574' without his specific permission. The RP manager also directed that a full survey of RW 574' be performed, including air samples. The access restriction was documented in the RP narrative log at 1114 hours as a standing order. No further entries into RW 574' elevation occurred after the RP manager issued the standing order.

The specific radiation work permit to survey RW 574' was generated 2 days later on Thursday, June 7, 2012. At 1514 hours, an RP technician entered RW 574' to perform a complete radiological survey of the area and determined that the conditions for a locked high radiation area were met. The RP technician stopped work and notified the on-duty RP supervisor. At this point, appropriate radiological postings and controls were established on RW 574'. Dose rates in the area were determined to be 2200 millirem/hour contact and 1600 millirem/hour at 30 centimeters from the 2 to 3 inches of resin on the floor outside the CBST tank room. The RW building elevator was also taken out of service with an RP lock installed as part of establishing the locked high radiation area boundary conditions to prevent access into the area.

<u>Analysis</u>: The inspectors concluded that the licensee's failure to perform radiological surveys and evaluate the potential radiological hazards in response to known degradation of the radiological and material conditions on RW building 574' was a performance deficiency consistent with IMC 0612 "Power Reactor Inspection Reports," dated July 10, 2012. The inspectors determined that the finding and each example therein, was more than minor because it was associated with the program and process attribute of the Occupational Radiation Safety cornerstone and adversely affected the cornerstone objective of protecting worker health and safety from exposure to radiation. Specifically, not barricading and conspicuously posting locked high radiation areas may result in unnecessary and unplanned radiation exposures to workers. The inspectors reviewed IMC 0612, Appendix E, "Examples of Minor Issues," dated August 11, 2009, and found no similar performance deficiencies. The inspectors concluded that each example occurred independently of the others and that each example provided the licensee with the opportunity to identify and correct the performance deficiency. The finding was not subject to traditional enforcement since the incidents did not have a significant safety consequence, did not impact the NRC's ability to perform its regulatory function, and were not willful. Additionally, the inspectors determined that the performance deficiency was reasonably within the licensee's ability to foresee and correct and was indicative of current performance.

In accordance with IMC 0609, Appendix C, "Occupational Radiation Safety Significance Determination Process," dated August 19, 2008, the inspectors determined that the finding had very low safety significance because the finding was not an ALARA (As-Low-As-Is-Reasonably-Achievable) planning issue, there was no overexposure or substantial potential for overexposure, and the licensee's ability to assess dose was not compromised. The licensee documented this issue in the CAP. Corrective actions included surveying, posting, barricading, and controlling access to the area.

The inspectors concluded that the most significant contributor to the finding was in the cross-cutting area of Human Performance with the component of decision making. The specific aspect was that the licensee uses conservative assumptions in decision making and adopts a requirement to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disprove the action (H.1(b)).

<u>Enforcement</u>: 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 10 CFR 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. Title 10 CFR 20.1003 defines survey as 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.

Title 10 CFR 20.1601 requires control for access to high radiation areas and subpart (c) allows a licensee to apply to the NRC for approval of alternative methods for controlling access to high radiation areas.

Technical Specification 5.7, which implements an NRC-approved alternate method for controlling access to high radiation areas, requires that the licensee appropriately barricade and conspicuously post an area that was accessible to personnel with radiation levels such that a major portion of the whole body could receive in 1 hour a dose greater than or equal to 1000 millirem.

Contrary to the above, from June 3 through June 7, 2012, the licensee did not perform surveys to assure compliance with 10 CFR 20.1601 and TS 5.7, which requires the control of access to high radiation areas. Specifically, on three separate occasions, the licensee failed to adequately evaluate the radiological conditions and potential hazards in areas on RW 574' that were accessible to personnel with radiation levels such that a major portion of the whole body could receive in 1 hour a dose greater than or equal to 1000 mRem, and failed to adequately barricade access to those areas.

Corrective actions included performing complete radiological surveys of the area, and posting and controlling the area as required by licensee TSs. These actions were completed on June 7, 2012. Because this violation is of very low safety significance and it was entered into the licensee's CAP (as CR 2012-09447), this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. (**NCV 05000440/2012005-02; Failure to Appropriately Control Access to a Locked High Radiation Area**)

.2 Instructions to Workers (02.03)

a. Inspection Scope

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

b. Findings

No findings were identified.

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

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

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

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

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

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

b. Findings

No findings were identified.

.4 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

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

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

- Radiation Work Permit 120134; HCA Strongback Cutting and Removal;
- Radiation Work Permit 120141; C510001E TIP Drive Activities; and
- Radiation Work Permit 120206; 0G50D0001 Waste Collector Filter Room Belt Replacement – High Radiation Area/Airborne.

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

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

b. Findings

No findings were identified.

.5 Radiation Worker Performance (02.07)

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.6 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

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

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

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Mitigating Systems, Barrier Integrity, Occupational Radiation Safety

4OA1 Performance Indicator Verification (71151)

.1 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the Reactor Coolant System (RCS) Specific Activity performance indicator (PI) for the third quarter 2011 through the third quarter 2012. To determine the accuracy of the PI data reported, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, was used. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, issue reports, event reports, and NRC integrated IRs to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a reactor coolant system sample. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RCS specific activity sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Exposure Control Effectiveness PI for the third quarter 2011 through the third quarter 2012. To determine the accuracy of the PI data reported, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, was used. The inspectors reviewed the licensee's assessment of the PI for occupational radiation safety to determine if indicator-related data was adequately assessed and reported. To assess the adequacy of the licensee's PI data collection and analyses, the inspectors discussed with RP staff the scope and breadth of their data review and the results of those reviews. The inspectors independently reviewed electronic personal dosimetry dose rate, and accumulated dose alarms, and dose reports, and the dose assignments for any intakes that occurred during the time period reviewed to determine if there were potentially unrecognized occurrences. The inspectors also conducted walkdowns of numerous locked high and very high radiation area entrances to determine the adequacy of the controls in place for these areas. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

Introduction: The inspectors opened an Unresolved Item (URI) pending the licensee's final data submittals for the occupational exposure control effectiveness PI for the second and third quarters of 2012.

Description: The licensee submitted annotated PI data for the second and third quarter of 2012. Specifically, the licensee will finalize the second quarter 2012 data after reviewing the NRC response to an NEI frequently asked question (FAQ 2012-004). The licensee will finalize the third quarter 2012 data after completion of the root cause evaluation of potential inadequate locked high radiation area controls for a loaded HI-STORM spent fuel storage container. The NRC will complete its review upon submittal of finalized data by the licensee. **(URI 05000440/2012005-07, Follow-Up to Occupational Radiation Safety Performance Indicator Verification)**

.3 Radiological Effluent Technical Specification/Offsite Dose Calculation Manual Radiological Effluent Occurrences

a. Inspection Scope

The inspectors sampled licensee submittals for the Radiological Effluent TS/Offsite Dose Calculation Manual (RETS/ODCM) Radiological Effluent Occurrence PI for the third quarter 2011 through the third quarter 2012. To determine the accuracy of the PI data reported, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, was used. The inspectors reviewed the licensee's issue report database and selected individual reports generated since this indicator was last reviewed to identify any potential occurrences such as unmonitored, uncontrolled, or improperly calculated effluent releases that may have impacted offsite dose. The inspectors reviewed gaseous effluent summary data and the results of associated offsite dose calculations for selected dates to determine if indicator results were accurately reported. The inspectors also reviewed the licensee's methods for quantifying gaseous and liquid effluents and determining effluent dose. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RETS/ODCM radiological effluent occurrences sample as defined in IP 71151-05.

b. Findings

No findings were identified.

.4 Reactor Coolant System Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage PI for the fourth quarter 2011 through the third quarter 2012. To determine the accuracy of the PI data reported, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports, and NRC integrated IRs to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one RCS leakage sample as defined in IP 71151-05.

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Annual Sample: Review of Operator Workarounds

a. Inspection Scope

The inspectors evaluated the licensee's implementation of the process used to identify, document, track, and resolve operational challenges. Inspection activities included, but were not limited to, a review of the cumulative effects of the operator workarounds (OWAs) on system availability and the potential for improper operation of the system, for potential impacts on multiple systems, and on the ability of operators to respond to plant transients or accidents.

The inspectors performed a review of the cumulative effects of OWAs. The documents listed in the Attachment to this report were reviewed to accomplish the objectives of the IP. The inspectors reviewed both current and historical operational challenge records to determine whether the licensee was identifying operator challenges at an appropriate threshold, had entered them into the CAP and proposed or implemented appropriate and timely corrective actions which addressed each issue. Reviews were conducted to determine if any operator challenge could increase the possibility of an Initiating Event, if the challenge was contrary to training, required a change from long-standing operational practices, or created the potential for inappropriate compensatory actions. Additionally, all temporary modifications were reviewed to identify any potential effect on the functionality of Mitigating Systems, impaired access to equipment, or required equipment uses for which the equipment was not designed. Daily plant and equipment status logs, degraded instrument logs, and operator aids or tools being used to compensate for material deficiencies were also assessed to identify any potential sources of unidentified OWAs.

This review constituted one OWA annual inspection sample as defined in IP 71152-05.

b. Findings

No findings were identified.

.4 Selected Issue Follow-Up Inspection: Actions to Upgrade/Improve Plant Under Drain System

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting the continuing problem of maintaining plant underdrain level within the required limits established in the licensee's USAR report. The inspectors performed an in-depth review of several CRs, including assigned corrective actions, and reviewed current plans to reduce groundwater levels with licensee personnel. The inspectors verified the following attributes during their review of the above CRs:

- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;

- consideration of previous occurrences;
- classification and prioritization of the resolution of the problem, commensurate with safety significance;
- identification of the contributing causes of the problem; and
- identification of corrective actions, which were appropriately focused to correct the problem.

The inspectors discussed the corrective action plans with licensee personnel.

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

b. Findings

No findings were identified.

.5 Selected Issue Follow-Up Inspection: Operator Decision Making/Technical Specification Evaluations

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting the decision making and TS evaluation process used by operators for a malfunctioning remote shutdown switch that caused a containment isolation valve indication to not indicate the current position of the isolation valve. The inspectors performed an in-depth review of CR 2012-14283, "RHR 'B' minimum flow valve lost indication when opened from the control room." Assigned corrective actions were reviewed with licensee personnel and the inspectors verified the following attributes during their review of the above CR:

- accurate and timely operability determination;
- complete and accurate identification of the problem in a timely manner commensurate with its safety significance and ease of discovery;
- consideration of previous occurrences;
- classification and prioritization of the resolution of the problem, commensurate with safety significance;
- identification of the contributing causes of the problem; and
- identification of corrective actions, which were appropriately focused to correct the problem.

The inspectors discussed the corrective action plans with licensee personnel.

This review constituted the second of two in-depth problem identification and resolution samples as defined in IP 71152-05.

b. Findings

No findings were identified.

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

Unusual Event Declared For Toxic Gas (Acetylene Gas Presenting Itself As Carbon Monoxide Gas) In The Radwaste Control Room At Levels Requiring Personnel To Leave The Radwaste Control Room

a. Inspection Scope

The inspectors reviewed the plant's response to an Unusual Event following increases in indicated carbon monoxide levels in the RW control room. Maintenance workers were preparing to perform welding on ventilation ducting in the semi-enclosed overhead above the RW control room and had taken an acetylene bottle into the overhead. Unknown to the workers, the acetylene bottle was leaking and the gas presented itself as carbon monoxide. When levels of "carbon monoxide" continued to rise and the source at the time was not readily apparent, the shift manager directed personnel in the RW control room to evacuate the control room, at which time the shift manager declared an Unusual Event in accordance with EAL MU1 – "Toxic Gas." The licensee remained in the unusual event until it was determined what the source of the toxic gas was, which was the leaking acetylene bottle, and after the area had been sufficiently ventilated and the atmospheric conditions had returned to safe levels for personnel occupancy. Documents reviewed in this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01)"

a. Inspection Scope

The inspectors verified that the onsite documentation, system hardware, and licensee actions were consistent with the information provided in the licensee's response to NRC Generic Letter (GL) 2008-01, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." Specifically, the inspectors verified that the licensee implemented or was in the process of implementing the commitments, modifications, and programmatically controlled actions described in the licensee's response to GL 2008-01. The inspection was conducted in accordance with Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01)," and considered the site-specific supplemental information provided by the Office of Nuclear Reactor Regulation (NRR) to the inspectors.

b. Inspection Documentation

The selected TI areas of inspection were licensing basis, design, testing, and corrective actions. The documentation of the inspection effort and any resulting observations are below.

Licensing Basis: The inspectors reviewed selected portions of licensing basis documents to verify consistency with the NRR assessment report. The licensing basis verification included the verification of selected portions of TSs, TS basis, and the USAR. The inspectors also verified applicable documents that described the plant and plant operation, such as calculations, piping and instrumentation diagrams, procedures, and CAP documents that addressed the areas of concern and were changed as needed. The inspectors also confirmed the frequency of selected surveillance procedures were at least as frequent as required by TSs. Finally, the inspectors verified the commitment to evaluate and implement the applicable changes that are contained in the TS Task Force traveler are consistent with the commitment described in NRR's Assessment Report and address any comments provided by NRR. The inspectors also conducted a licensing basis verification in an earlier inspection period associated with a corrective action item, documenting an air void observed in the RHR 'C' LPCI piping. This was identified by the licensee under CR 11-90863. This additional activity counted towards the completion of this TI and was documented in NRC IR 05000440/2011003.

Design: The inspectors reviewed selected design documents, performed system walkdowns, and interviewed plant personnel to verify that design and operating characteristics were addressed by the licensee. Specifically:

- The inspectors verified the licensee had identified the gas intrusion mechanisms that apply to the licensee's plant. If the licensee's evaluation was incomplete, the inspectors verified that corrective actions were placed into the CAP.

  Based on the initial review by the inspectors it appeared that the PNPP had not considered High – Low Pressure interface as a credible gas intrusion mechanism. Further review and discussion with the licensee determined that the problem was actually a lack of sufficient detail and rigor in the GL 2008-01 evaluations submitted to the NRC. The inspectors discussed with the licensee the potential negative impact this lack of detail and rigor could have on future inspections and licensee evaluations. This issue is compounded by the lack of an overall gas management program document. As a result, the licensee will determine the appropriate way to amend or provide additional information on the docket to clarify this discrepancy and has agreed to develop a program document for gas managing. This issue was captured under CR 2012-14509.

  As mentioned above, the lack of a gas management program document was discussed with the licensee. Proper implementation and understanding of all the required aspects of GL 2008-01 are essential to a successful program. For the purpose of the inspection, the licensee was able to provide the necessary information. However, on occasion the licensee needed to reconstitute certain information which had not been properly documented or explained under existing documentation. The licensee was able to reconstitute the missing information by interviewing the individuals originally involved. However, having to depend on being able to track down and interview the originally involved individuals was of concern to the inspectors. In particular, there is no guarantee these or other individuals would always be available. Although not a requirement, the lack of a central program document and/or program owner is a potential error trap. This issue was re-emphasized by the inspectors; however, the licensee had already identified it during their self-assessment. This issue was captured under CR 2012-14244. In addition, the licensee provided coaching to certain members of the plant staff on the site's expectation for the level of documentation required while performing gas quantification evaluations.

- The inspectors verified the licensee's void acceptance criteria were consistent with NRR's void acceptance criteria. If NRR's acceptance criteria were not met, then the inspectors verified that the licensee had justified the deviations. The inspectors also confirmed the range of flow conditions evaluated by the licensee was consistent with the full range of design basis and expected flow rates for various break sizes and location.

  The inspectors provided two observations where the licensee's void acceptance criteria (GEN-019, "ECCS [emergency core cooling system] Void Acceptable Criteria") were not consistent with NRR's criteria.

  (a) The licensee's acceptance criteria did not account for all the flow conditions as described in the guidance. The guidance provides different maximum acceptable voids fractions depending on a number of specified conditions. One of those conditions is based on what is the actual flow (Q) though the system in relation to the flow at the best efficiency point (Q_{bep}). Specifically, the licensee did not account for flow conditions where the $\frac{Q}{Q_{BEP}}\% < 70\ percent.$ Under these conditions the acceptable void fraction (under transient conditions) is more limiting. After discussing this issue, the licensee demonstrated via calculation that based on their design and configuration, they would not enter the above-mentioned flow conditions while having a Froude number high enough where a void would transport to the pump's suction. The licensee documented and tracked this issue under CR 2012-14311; and

  (b) As part of GEN-019, the licensee developed acceptance criteria for the allowable quantities of gas in the ECCS suction and discharge sides. The basis for the suction side acceptance criteria assumed a constant void fraction throughout the transient time. However, when calculating the allowable void volumes, the licensee did not account for potential slug flow. If slug flow occurred, the acceptable void fractions could have been exceeded. Therefore, the inspectors determined the licensee failed to assure that applicable regulatory requirements and the design basis were correctly translated into specifications, drawings, procedures, and instructions. This was contrary to 10 CFR Part 50, Appendix B, Criterion III, "Design Control," and was a performance deficiency. The performance deficiency was determined to be of minor significance, because further review of the acceptance criteria determined the existing criteria values were unaffected by this issue and the inspectors did not find an example where a void's as-found volume questioned the system's operability. This issue was captured under CR 2012-14311 and CR 2012-15564.

- The inspectors selectively reviewed applicable documents, including calculation and engineering evaluations, with respect to gas accumulation in the subject systems. Specifically, the inspectors verified these documents addressed venting requirements, keep-full systems, void control during system realignments, and the effect of debris on strainers in containment emergency sumps causing accumulation of gas under the upper elevation of strainers and the impact on net positive suction head requirements.

- The inspectors conducted a walkdown of selected regions of high-pressure core spray (HPCS). The inspectors also selectively verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspectors' independent walkdown.

- In addition, the inspectors verified the licensee had piping and instrumentation diagrams, and isometric drawings that describe the low-pressure core spray (LPCS) system's configurations and had confirmed the accuracy of the drawing's resolution. The inspectors verified the following related to the isometric drawings:

  (a) high point vents were identified;
  (b) high points that do not have vents were acceptably recognizable;
  (c) other areas where gas can accumulate and potentially impact subject system operability, such as at orifices in horizontal pipes, isolated branch lines, heat exchangers, improperly sloped piping, and under closed valves, were acceptably described in the drawings or in referenced documentation;
  (d) horizontal pipe centerline elevation deviations and pipe slopes in nominally horizontal lines that exceed specified criteria were identified;
  (e) all pipes and fittings were clearly shown; and
  (f) the drawings were up-to-date with respect to recent hardware changes and any discrepancies between as-built configurations and the drawings were documented and entered into the CAP for resolution.

- The inspectors conducted a walkdown of selected portions of LPCS in an earlier inspection period. This additional activity counted towards the completion of this TI and was documented in NRC IR 05000440/2011003.

- The inspectors verified the licensee's walkdowns have been completed.

Testing: The inspectors reviewed selected surveillances, post-modification tests, and post-maintenance test procedures, and results, to verify the licensee had approved and was using procedures that were adequate to address the issue of gas accumulation and/or intrusion in the subject systems. This review included the verification of procedures used for conducting surveillances and determination of void volumes to ensure the void criteria were satisfied and will be reasonably ensured to be satisfied until the next scheduled void surveillance. Also, the inspectors reviewed procedures used for filling and venting following conditions which may have introduced voids into the subject systems to verify the procedures addressed testing for such voids and provided processes for their reduction or elimination. The inspectors also reviewed selected portions of procedures used during the surveillance of HPCS pump and valve operability testing in an earlier inspection period. This additional activity counted towards the completion of this TI and was documented in NRC IR 05000440/2010005.

Corrective Actions: The inspectors reviewed selected licensee assessment reports and CAP documents to assess the effectiveness of the licensee's CAP when addressing the issues associated with GL 2008-01.

The documents reviewed are listed in the Attachment to this report.

Based on this review, the inspectors concluded that there is reasonable assurance that the licensee will complete all outstanding items and incorporate this information into the design basis and operational practices. Therefore, this TI is considered closed

c. Findings

(1) Inappropriate Procedures For Restoring Low-Pressure Coolant Injection Mode of Residual Heat Removal Following A Loss-of-Coolant Accident at Mode 3

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions Procedures and Drawings," was identified by the inspectors for the failure to establish appropriate procedures capable of restoring LPCI mode of RHR, while in the SDC mode, following a LOCA at Mode 3.

Description: On June 16, 2010, the NRC issued Information Notice (IN) 2010-11, "Potential for Steam Voiding Causing Residual Heat Removal System Inoperability." The IN discussed issues identified at a number of operating nuclear power plants where on multiple occasions, the RHR systems were inoperable because of the potential for steam voids at the RHR pump suction piping.

Perry's TS 3.5.1, "ECCS-Operating," requires each ECCS injection/spray subsystem and the automatic depressurization system function of eight safety/relief valves shall be operable. The TS is applicable in Modes 1, 2, and 3. The bases of TS 3.5.1 state:

> "LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal (DHR) when below the actual RHR cut in permissive pressure in Mode 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable."

As part of the automatic actions following a design basis LOCA, SDC would isolate and in the process trap water at temperatures above saturation. Based on discussions with the licensee, and on the RHR cut-in permissive setpoint; water at a temperature as high as 358 $^{\circ}$F could be present in the RHR system following a LOCA during Mode 3. This presents a challenge when re-aligning the RHR system from SDC to LPCI, since the reactor would be at a significantly lower pressure. The re-alignment of the RHR system would cause the high temperature and pressure water to flash to steam. This introduces a concern where the RHR pump could become air bound and/or that water hammer events could occur. In order to meet T.S. 3.5.1 and maintain all LPCI subsystems operable while any LPCI subsystem is aligned to SDC, the licensee needed to prove that appropriate actions could be taken to restore any LPCI train aligned to SDC following a postulated design basis LOCA during Mode 3.

As part of their TI-177 inspection effort, the inspectors requested the licensee's evaluation of the above-mentioned IN. The licensee had evaluated IN 2010-11 under Work Order 200421560. While reviewing the evaluation, the inspectors noted the licensee had identified the potential for steam voiding and water hammer events to occur during a system lineup change (from SDC to LPCI) with elevated system temperatures at the pump suction. The licensee's evaluation concluded that "sufficient barriers are in place to prevent the occurrence of steam voiding in the [Perry] RHR system." However, the inspectors identified a number of deficiencies, because if left uncorrected, would have the potential to lead to more significant safety concerns. Specifically, the current procedures used by the licensee to restore LPCI mode in the affected train:

- depended on the use of nonsafety-related SSCs for restoring LPCI (e.g., condensate transfer pumps);

- Off-Normal Instructions (ONIs) SPI-A9 and SPI-B9 instructed operators to inject relatively cold water into a potentially voided line. This activity had the potential to cause a water hammer event in the system, which had not been evaluated;

- procedures required the use of hoses in order to fill and vent the system following the event. However, these hoses were only rated for temperatures of about 190 °F. The steam from the RHR system could be as hot as 358 °F;

- Current procedures had no warnings or cautions to the operator that hot water/steam could be exiting the vents. As a result, proper industrial and radiological safety measured might not have been taken. The vented system was part of the RCS;

- The licensee captured the inspectors' concerns on CR 2012-15550 and CR 2012-14830. In addition, the licensee implemented a number of compensatory actions in the interim;

- The licensee issued an operation standing order dated October 4, 2012. The order requires the LPCI mode of RHR be declared INOPERABLE when the associated loop is running SDC with the plant in Mode 3; and

- The licensee issued an instruction to ensure that procedures ONI-SPI A-9 and ONI-SPI B-9 (both Revision 1) are not used when the plant is in Mode 3.

The corrective actions being considered at the time of this inspection included evaluating for alternate strategies to mitigate the accident and revising the affected procedures as required.

Analysis: The inspectors determined the failure to establish appropriate procedures capable of restoring LPCI mode of RHR, while on SDC mode, following a LOCA at Mode 3 was contrary to 10 CFR Part 50, Appendix B, Criterion V, "Instructions Procedures and Drawings," and was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, because, if left uncorrected, it would have the potential to lead to a more significant safety concern. Specifically, the inspectors had concerns that procedures, as currently written, would have been unsuccessful in restoring LPCI. In addition, if operators would have used the current procedures, there was the potential to cause unanalyzed water hammer events in the RHR system, and the equipment listed as required might not have been adequately rated for its use and/or could have exposed the operator to dangerous/unsafe industrial and radiological environments.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," dated June 19, 2012. Because the finding was associated with shutdown conditions, the inspectors used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," Checklist 5, "BWR Hot Shutdown: Time to Boil < 2 Hours RHR in Operation," dated February 28, 2005, to evaluate the significance of the finding. The finding required a Phase II SDP evaluation because it degraded the licensee's ability to add RCS inventory when needed since the RHR train in operation would fail if realigned from the SDC mode to the RCS injection mode. The inspector determined the exposure time for this finding was very short. The plant had operated in this condition for a total of 12.5 hours since 2009.

The Region III Senior Reactor Analyst performed a Phase II SDP evaluation using IMC 0609, Appendix G, Attachment 3, Worksheet 1, and "SDP Worksheet for a BWR Plant – Loss of Inventory in POS 1 (Head On)." The Senior Reactor Analyst determined that the manual low-pressure injection function was potentially affected by the finding. However, the SRA determined that since one full train of RHR and the LPCS system were unaffected, credit for manual injection would remain unchanged. The initiating event frequency was determined to be a "4" given the short exposure period. The affected sequences were solved and the result was a delta core damage frequency (CDF) less than 1.0E-6/year, a finding of very low safety significance (Green). The dominant sequence was a loss of inventory followed by a failure of manual injection of low and high-pressure injection systems.

The inspectors determined this finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because the licensee did not implement operating experience through changes to the station's process, procedures, and equipment. Specifically, the licensee's evaluation of IN 2010-11 incorrectly concluded sufficient barriers were in place to prevent the occurrence of steam voiding in the RHR system." [P.2(b)]

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

Contrary to the above, as of October 5, 2012, the licensee did not have procedures appropriate for restoring LPCI mode of RHR following a Mode 3 LOCA. Specifically, the licensee failed to prescribe procedures which ensured that LPCI could be restored using only safety-related/seismic SSCs, ensured that no unanalyzed water hammer event occurred, ensured that the equipment (hoses) used for venting the system were appropriate and that operator safety was maintained. As part of the corrective actions, the licensee started compensatory action to declare any train of RHR inoperable while aligned to SDC and to prohibit the use of ONI-SPI A-9 and ONI-SPI B-9 while the plant is in Mode 3, until an adequate strategy for restoring LPCI following a Mode 3 LOCA could be developed. Because this violation was of very low safety significance, and was entered into the licensee's CAP as CR 2012-14830 and CR 2012-15550, this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (**NCV 05000440/2012005-03, Inappropriate Procedures for Restoring LPCI Mode of RHR Following a LOCA at Mode 3**)

(2) Deficiencies With Periodic Venting Procedures and Void Quantification

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," was identified by the inspectors for the failure to ensure adequate test instrumentation was available and used during the performance of periodic venting.

Description: On January 11, 2008, the NRC requested each addressee of GL 2008-01 to evaluate its ECCS, decay heat removal (DHR), and containment spray systems licensing basis, design, testing, and corrective actions to ensure that gas accumulation was maintained less than the amount that would challenge the operability of these systems, and take appropriate actions when conditions adverse to quality were identified. As part of the action in response to GL 2008-01, the licensee identified the locations where periodic monitoring for voids was required.

One of the guidance documents used by the licensee in developing its gas management strategy was NEI 09-10, Revision 0, "Guidelines for Effective Prevention and Management of System Gas Accumulation." The NEI guidance and the guidance provided in GL 2008-01 discussed the necessity to quantify gas identified in the subject systems. In order to accomplish void quantification, different methods were suggested. The method chosen by the licensee required operators to "crack open" the vent valves and estimate the time necessary to vent the pipe or component. The guidance recognized that when this method is used, care should be taken when using the results, as the variability in the method could be significant. In addition, the guidance states that when using this method, the degree of accuracy required at a given location should be evaluated and documented.

The inspectors reviewed samples of the procedures used during periodic venting and results of recent venting performance. The inspectors observed that SVI E12-T1182-A, Revision 12, "RHR A LPCI Valve Lineup Verification and System Venting," required operators to estimate the time air was observed during the venting process. However, it did not require operators to use a stopwatch to measure said time. The inspectors were concerned because the amount of time recorded was used to quantify the volume of the void, if any was found. By not requiring operators to use a stopwatch, or other timepiece, the licensee introduced more uncertainty into an already inaccurate method. This deficiency was present on all other ECCS SVIs.

The inspector reviewed previous CRs in order to determine how the recorded times were used to quantify the gas volumes, how were the method's uncertainty bounded, and to determine how sensitive the results were to timing errors. The licensee provided examples where air was identified during periodic venting. The inspectors reviewed these examples (CR 2011-90863, CR 2011-96588 and CR 2011-95230). These CRs did not document the process used by the licensee to quantify the amount of gas found during the venting process. As a result, the inspectors were unable to verify the licensee's method and establish why it was acceptable to estimate time of venting without the use of a timepiece. The licensee referenced "conservative calculations" or "conservative estimates" used to determine the void volume, but no bases for these statements were provided. For one of the cited examples, CR 2012-95230, the licensee's evaluation assumed choke flow and a fully opened vent valve. Based on the calculated exiting volumetric flow rate, the licensee would have exceeded its acceptance criteria if any air was observed for approximately 3 to 5 seconds. It is prudent to note that these results were only applicable to the specific venting location mentioned on the CR. However, this was evidence that for some locations, the difference between an acceptable void and one that could challenge the operability of the system, would be a matter of seconds. The licensee documented the inspectors' concern as CR 2012-14487 and agreed to revise their periodic venting procedures to require the use of a stopwatch.

Another weakness noted with the procedures was the lack of guidance to the operators when instructing them to record the amount of time air was vented from the system. The SVIs did not instruct operators to estimate the amount of air vented until the end of the procedure. In theory, an operator was not instructed to record or remember the amount of time gas was vented until after all the venting locations had been performed. This could result in relying on the memory of the operator to remember (an estimate) of the time air vented from each location. In practice, the expectations on operators was that if anything more than a second of air was observed they would stop and contact the control room. At that time, the control room should ask the operators for an estimate of time air was vented. The inspectors believed this presented an additional error trap. The licensee also documented this issue as part of CR 2012-14487.

Analysis: The inspectors determined the failure to ensure that adequate test instrumentation was available and used during performance of periodic venting was contrary to 10 CFR Part 50, Appendix B, Criterion XI, and was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with IMC 0612, Appendix B, "Issue Screening" dated September 7, 2012 because, it was associated with the Mitigating Systems cornerstone and adversely affected the cornerstone attribute of "Procedure Quality: Maintenance and Testing Procedures." Specifically, by not using adequate test instrumentation, e.g., a timepiece, to measure the time gas was vented, it introduces further uncertainty to an already inaccurate method. This was important, in particular, when considering (1) the examples where acceptable void volume could be exceeded in a matter of seconds; and (2) there were no documented examples where the licensee could show how the uncertainty and unknowns associated with the method used were being bounded and/or accounted.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Initial Characterization of Findings," dated June 19, 2012. Because the finding impacted the Mitigating Systems cornerstone, the inspectors screened the finding through IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," using Exhibit 2, "Mitigating Systems Screening Questions," dated June 19, 2012. The finding screened as very low safety significance (Green) because the finding was a qualification deficiency confirmed not to result in loss of operability or functionality. Specifically, review of the licensee's CAP documents since the implementation of the resolution of GL 2008-01 determined that voids had been identified following system restoration (initial fill and vent) when the system was inoperable, and voids identified when the system was online had been significantly below the calculated acceptance criteria. In addition, discussion with the licensee determined that voids found while the system was online were in dead leg pipe segments and/or outside the main system flowpath.

The inspectors determined this finding had a cross-cutting aspect in the area of Problem Identification and Resolution, Operating Experience, because the licensee did not thoroughly evaluate relevant external operating experience. Specifically, licensee evaluation of NEI 09-10 Revision 0, failed to identify the importance of having adequate venting time information when quantifying vented voids (P.2(a)).

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion XI, "Test Control," required in part, that test procedures include provisions for assuring that all pre-requisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

Contrary to the above, as of October 5, 2012, the licensee failed to ensure adequate test instrumentation was available and used. Specifically, the licensee's test procedures failed to require the use of a timepiece (i.e., stopwatch) when recording the venting times as part of the periodic SVI. As part of the planned corrective actions, the licensee planned to revise the affected procedures to include the use of a stopwatch. Because this violation was of very low safety significance and it was entered into the licensee's CAP (as CR 2012-14487), this violation is being treated as an NCV, consistent with Section 2.3.2 of the NRC Enforcement Policy. (**NCV 05000440/2012005-04, Deficiencies with Periodic Venting Procedures and Void Quantification**)

.2 (Closed) TI 2515/185, Revision 1, Follow-Up On The Industry's Groundwater Protection Initiative

a. Inspection Scope

An NRC assessment was performed on the Groundwater Protection Program to determine whether Perry station fully implemented the voluntary industry groundwater protection initiative, NEI 07-07, Industry Groundwater Protection Initiative (GPI) – Final Guidance, dated August 2007, ADAMS Accession Numbers ML072610036 and ML072600292. The inspectors interviewed personnel, reviewed applicable documents, and performed walkdowns of selected areas. In addition, the inspectors followed-up on the status of implementation for the five deviations to the acceptance criteria in NEI 07-07 that were documented in the NRC Integrated IR 05000440/2010002, dated May 4, 2010.

Specifically the inspectors reviewed NEI – GPI Objectives

- GPI Objective 1.1e - Site Hydrology and Geology.

  Perform a site characterization of the geology and hydrology that provides an understanding of the predominant ground water gradients based upon current site conditions. Revise the USAR to include changes to the characterization of hydrology and/or geology, as appropriate.

- GPI Objective 1.2b – Site Risk-Assessment.

Identify SSCs that involve or could reasonably be expected to involve licensed material for which there is a credible mechanism for licensed material to reach ground water. Identify leak detection methods for each SSC for which there is a credible mechanism for licensed material to reach ground water.

- GPI Objective 1.3f - Onsite Ground Water Monitoring

  Establish an onsite ground water monitoring program to ensure timely detection of inadvertent radiological releases to ground water including a long-term program for preventative maintenance of ground water wells.

- GPI Objective 1.4b - Remediation Process

  Evaluate the potential for detectible levels of licensed material resulting from planned releases of liquids and/or airborne materials.

- GPI Objective 2.2a - Voluntary Communications

  Develop guidance for voluntary communication to satisfy the thresholds provided in the NEI Initiative or state/local agreements.

b. Findings and Observations

No findings were identified.

.3 Initial Loading Campaign - Operation of an Independent Spent Fuel Storage Installation (ISFSI) at Operating Plants (60855.1)

a. Inspection Scope

The inspectors observed and evaluated the licensee's performance during loading the first and sixth canisters of the initial spent fuel storage campaign to verify compliance with the applicable Certificate of Compliance and TSs; 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste"; and associated procedures.

The inspectors observed heavy loads movements inside the fuel handling building (FHB) including: lifting the transfer cask (HI-TRAC) and placing it into the spent fuel pool (SFP); lifting the HI-TRAC from the SFP and placing it in the decontamination area; lifting the HI-TRAC from the decontamination area and placing it atop a storage cask (HI-STORM); and transfer of the multi-purpose canister (MPC) from the HI-TRAC to the HI-STORM while the casks were stacked on one another in a restrained configuration. The inspectors observed loading of spent fuel assemblies from the SFP into the MPC. The inspectors observed MPC processing operations including: decontamination and surveying; MPC welding; non-destructive weld examinations; vacuum drying; and helium backfilling.

During performance of the activities, the inspectors evaluated: the familiarity of the licensee's staff with procedures; supervisory oversight; and communication and coordination between the groups involved. The inspectors reviewed loading and monitoring procedures and evaluated the licensee's adherence to these procedures.

The inspectors verified that contamination and radiation levels of the HI-TRAC and HI-STORM were below the regulatory, TS, and administrative limits.

The inspectors reviewed CRs and the associated follow-up actions that were generated during the loading campaign. Specifically, the inspectors reviewed several CRs involving the FHB crane (CR 2012-12933 and CR 2012-13234) and questions that arose during welding operations on the sixth canister (CR 2012-18602). The inspectors also reviewed the licensee's 10 CFR 72.48, "Changes, Tests, and Experiments," screenings.

The inspectors also observed the licensee implement contingency procedures, following helium backfill, when a port cap did not seal in preparation for final closure of the MPC. The inspectors reviewed repair work orders, procedures, and regulatory reviews in support of replacing the port cap and were present throughout the actual repair. In addition, the inspectors followed investigations and corrective actions associated with abnormal noises and sparking coming from the vicinity of the FHB crane during movement of a load that did not contain spent nuclear fuel.

b. Findings

(1) Inadequate Fuel Handling Building Crane Maintenance Challenges Single-Failure-Proof Compliance

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed for failure to perform adequate maintenance on the single-failure-proof FHB crane used to handle dry storage casks containing spent nuclear fuel.

Description: The process of safely moving spent nuclear fuel from the SFP into dry storage relies on the single-failure-proof overhead crane located in the FHB. The function of the FHB, as stated in USAR Section 3.8.4.1.3.1, is "to store new fuel and to receive and store spent fuel." The USAR Table 3.2-1 classified the FHB crane (refueling cask crane) as Seismic Category I and Safety Class 3. Section 3.2.3.3.1 of the Perry USAR defines Safety Class 3 SSCs, in part, as those "whose function is to process radioactive wastes and whose failure would result in release to the environment of gas, liquid, or solids resulting in a single event dose greater than the limits specified in 10 CFR 100 to a person at the site boundary." The USAR Section 3.2.4 states that SSCs whose safety functions require conformance to the quality assurance requirements of 10 CFR Part 50, Appendix B, are those specified as Safety Class 1, 2, and 3 in Table 3.2-1. The FHB crane was installed in 1981 and was upgraded through Engineering Change Package (ECP) 04-0278-001, "FHB Crane Upgrade," Revision 0, to a 125-ton single-failure-proof compliant crane in accordance with NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The ECP 04-0278-001 states "the design and installation work for the FHB crane upgrade is classified as Nuclear Safety Related."

By design, the overhead crane cannot traverse the spent fuel assemblies located in the SFP. However, attached to the SFP is a cask pit that permits fuel assemblies to be moved from the SFP into a transfer cask located underwater in the cask pit. After a lid is set on the transfer cask, the FHB crane is used to lift the transfer cask out of the cask pit and placed in an area to permit processing for storage on the ISFSI pad.

The inspectors were onsite observing the lowering of an empty HI-TRAC into the cask pit when an abnormal noise and sparks were noted above near the vicinity of the crane. The load was stopped; however, after some consideration the licensee ultimately decided to lower the HI-TRAC to the bottom of the cask pit rather than leaving the load suspended. Upon further investigation, it was determined that the Magnetorque, a component that provides a power braking function, was the source of the sparks and appeared excessively hot. The Magnetorque is an eddy current load brake designed to provide a retarding torque during raising and lowering loads. During normal operation, the Magnetorque works in conjunction with the normal speed drive motor such that at slow hoisting speeds, the Magnetorque carries the majority of the load, and at normal speed the drive motor carries the majority of the load. This is accomplished through ammeter modules that change the current to each component (Magnetorque and normal drive motor) based on the hoisting speed. Once the investigation was complete, the current sent to the Magnetorque was identified to be significantly higher than it should have been. At the time of the crane malfunction, the hoist was operating at relatively slow speeds. The Magnetorque is cooled by an internal shaft-mounted fan that draws air in through vents. Due to excessive current supplied to the Magnetorque's rotor, combined with relatively low cooling provided by slow speed operations, the rotor temperature increased causing thermal expansion of the rotor. Because of small clearances between the rotor and stator, the rotor expanded to the point where contact occurred with the stator causing the abnormal noise and sparking.

NUREG-0554 requires that means should be provided to repair, adjust, or replace failed components in place with the load supported and retained in a safe position. As an alternative to repairing in place, an immobilized crane with a suspended load shall be provided with a means to transfer the load to a safe laydown area before implementing repairs. Calculation No. G58-S-R-L-006, Revision 0, section 3.4, "Emergency Repairs," describes that the Magnetorque is used, during abnormal operations, to satisfy NUREG-0554 requirements as follows, "the crane was modified to include an 'alternator' for the eddy current control brake [Magnetorque], a lowering speed display to facilitate the safe manual lowering of loads, and provided manual capability for positioning the bridge and trolley."

The control system was originally setup on May 28, 2010, by the vendor following the upgrade through ECP 04-0278-001. No other adjustments to these settings occurred prior to the lift when sparking was observed. Preventive Maintenance Instruction (PMI)-0031, "Fuel Handling Area Crane Preventive Maintenance," dated May 21, 2010, conducts inspections and testing of the crane. However, there were no provisions in PMI-0031 for the calibration of the static stepless controls. The vendor maintenance manual, 0180-021K, "Static Stepless Control," states in part, "it is good practice, however, to perform a complete setup procedure once a year to ensure continued optimum performance from the system." Contrary to the vendor's operating manual recommendations, the licensee had not performed annual checks of these currents.

The licensee entered this issue into the CAP (CR 2012-13234, CR 2012-13315, and CR 2012-12933). The crane vendor performed the recommended setup and evaluated the Magnetorque and other parts of the crane for potential damage.

Analysis: The inspectors determined that the failure to follow the vendor's recommended maintenance was contrary to the instructions, procedures, and drawings provisions of 10 CFR Part 50, Appendix B, Criterion V requirements, and was a performance deficiency. The inspectors determined the performance deficiency was more than minor in accordance with IMC 0612, Appendix B, "Issue Screening," dated September 7, 2012, in that it was associated with the Human Performance attribute (maintenance performance) of the Barrier Integrity cornerstone and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radioactive releases caused by accidents or events. Additionally, if left uncorrected, a malfunction of the FHB crane could lead to a more significant safety concern. Specifically, the single-failure-proof FHB crane is relied upon to safely handle spent nuclear fuel during dry cask operations over the cask pit pool. The inadequate maintenance challenged the functionality of the Magnetorque, a component that is credited in the design basis of the crane's single-failure-proof qualification.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process." Based on answering "No" to all the screening questions in IMC 0609, Appendix A, Exhibit 3, "Barrier Integrity Screening Questions," dated June 19, 2012, the finding was determined to be of very low safety-significance (Green).

The inspectors identified a Human Performance, Resources (H.2(c)) cross-cutting aspect associated with this finding in that the licensee failed to have complete, accurate, and up-to-date procedures that ensured personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee failed to have maintenance procedures that ensured the FHB crane would be capable of performing its single-failure-proof design functions that assure nuclear safety.

Enforcement: Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," states, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on May 21, 2010, PMI-0031, "Fuel Handling Area Crane Preventive Maintenance," was approved for conducting maintenance activities on the FHB crane; however, PMI-0031 did not contain the appropriate vendor recommended maintenance practices that affected the quality of a Safety Class 3 component.

Because this violation was of very low safety significance (Green) and it was entered into the licensee's CAP (as CR 2012-13234, CR 2012-13315, and CR 2012-12933), this violation is being treated as an NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000440/2012005-05; 072000069/2012002-01; Inadequate Fuel Handling Building Crane Maintenance Challenges Single-Failure-Proof Compliance)**

(2) Failure to Follow Procedures That Ensure Safe Movement of a Dry Fuel Storage Canister

Introduction: The inspectors identified a Severity Level (SL) IV NCV of very low safety significance of 10 CFR 72.150, "Instructions, Procedures, and Drawings," for the failure of the licensee to follow procedures that ensured the safe loading of a dry fuel storage canister.

Description: On September 27, 2012, the licensee was moving a loaded HI-STORM outside the FHB, for placement on the ISFSI pad, in accordance with General Maintenance Instruction (GMI)-0221, "HI-STORM Movement with Stack-up Seismic Restraint Installed," Revision 8. The procedure contains two caution statements during movement of the Zero Profile Transporter (ZPT) and HI-STORM. The first states that no object shall be within 12 inches, north and south, of the ZPT when it is set down. The second requires a minimum of 12 inches of track remain clear to the rollers of the ZPT. Once outside the building, movement of the ZPT and HI-STORM was stopped for the installation of the HI-STORM lid when the ZPT was close to the end of the rails. The inspectors noted that the remaining track on the south side of the ZPT did not appear to have 12 inches of track remaining clear to the rollers of the ZPT. Step 5.4.18, which immediately follows the Caution statements, states to "CONFIRM 12 inches of track remain clear on south side of ZPT." The inspectors questioned the licensee to determine if they completed Step 5.4.18 and if so, to demonstrate that 12 inches remain clear. Step 5.4.18 was already marked as complete on the working copy of the procedure. After questioning, the licensee measured the distance of clear track on the south side of the ZPT. The ZPT was approximately 7 inches from the end of the rail. The licensee failed to follow procedure GMI-0221, Step 5.4.18 to ensure 12 inches of track remained clear on the south side of the ZPT.

In preparation for the initial ISFSI loading campaign, the licensee performed a seismic stability analysis of the ZPT with a loaded HI-STORM, which is documented in Calculation G-58-H-HI-2115007, Revision 1. This calculation determined that during a seismic event, the maximum displacement of the ZPT along the rails is 11.25 inches. To ensure the ZPT would have adequate clearance during a seismic event, a 12-inch minimum distance of clear track, on the south side of the ZPT, was required in order to meet the assumptions of the seismic stability analysis.

Approximately 2 hours after the inspectors questioned the licensee in meeting the 12-inch requirement, the licensee moved the ZPT and loaded HI-STORM back within the requirements of the procedure. Therefore, the licensee was in an unanalyzed condition for approximately 2 hours. The licensee evaluated this condition as part of CR 2012-15087.

Analysis: The inspectors determined that the failure to follow procedures was contrary to the instructions, procedures, and drawings provisions of 10 CFR 72.150 requirements that warranted a significance evaluation. Consistent with the guidance in Section 2.2 of the NRC Enforcement Manual, ISFSIs are not subject to the SDP, and thus, traditional enforcement will be used for these facilities. Therefore, the violation was dispositioned using the traditional enforcement process of Section 2.3 of the Enforcement Policy.

The violation was determined to be of more than minor significance using IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," dated

August 11, 2009, Example 2f, in that a procedural requirement was not met and the actual distance between the HI-STORM and the end of the rails was less than the analyzed distance required to ensure safe transport operations.

Consistent with the guidance in Section 2.6.D of the NRC Enforcement Manual, if a violation does not fit an example in the Enforcement Policy Violation Examples, it should be assigned a severity level: (1) Commensurate with its safety significance; and (2) informed by similar violations addressed in the Violation Examples. The inspectors determined that the violation could be evaluated using section 6.5.d.3 of the NRC Enforcement Policy (dated June 7, 2012), as a SL IV violation, in that the licensee failed to follow procedures affecting the safe transport of a loaded HI-STORM.

Cross-cutting aspects are not assigned to traditional enforcement violations. Since this violation was dispositioned using traditional enforcement, a cross-cutting aspect is not applicable.

Enforcement: Title 10 CFR Part 72.150, "Instructions, Procedures, and Drawings," requires, in part, that the licensee prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed.

Contrary to the above, on September 27, 2012, the licensee failed to follow a procedural requirement, for approximately 2 hours, that the actual distance between the HI-STORM and the end of the rails was greater than the analyzed distance required to ensure safe transport operations. After approximately 2 hours, the licensee moved the ZPT and HI-STORM to within the analyzed limits and entered the issue into the CAP (CR 2012-15087). This violation is being treated as an NCV, consistent with Section 2.3.2 of the Enforcement Policy because it was of very low safety significance (SL IV) and has been documented in the licensee's CAP. (**SLIV 05000440/2012005-06; 07200069/2012002-02**; **Failure to Follow Procedures That Ensure Safe Movement of a Dry Fuel Storage Canister**)

.4. (Closed) NRC TI 2515/188 – Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns

a. Inspection Scope

The inspectors accompanied the licensee on a sampling basis during the seismic walkdowns on August 10, 2012, on the Auxiliary Building 574', 599', and 620' elevations. The inspectors verified that the licensee conducted the walkdown activities using the methodology endorsed by the NRC. These walkdowns were performed at all operating reactor sites in response to a letter from the NRC to licensees, entitled "Request for Information Pursuant to Title 10 of the *Code of Federal Regualtions* 50.54(f) Regarding Recommendations 2.1, 2.3, and 9.3, of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," dated March 12, 2012 (ADAMS Accession No. ML12053A340). The inspectors confirmed that the following seismic features associated with HPCS motor-driven pump, HPCS injection valve, and HPCS flow transmitter rack were free of potential adverse seismic conditions:

- anchorage was free of bent, broken, missing or loose hardware;
- anchorage was free of corrosion that is more than mild surface oxidation;

- anchorage was free of visible cracks in the concrete near the anchors;
- anchorage configuration was consistent with plant documentation;
- SSCs will not be damaged from impact by nearby equipment or structures;
- overhead equipment, distribution systems, ceiling tiles, and lighting are secure and not likely to collapse onto the equipment;
- attached lines have adequate flexibility to avoid damage;
- the area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area;
- the area appears to be free of potentially adverse seismic interactions that could cause a fire in the area; and
- the area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment, and temporary installations (e.g., scaffolding, lead shielding).

The inspectors independently performed walkdowns in December 2012 and verified that the following seismic features associated with standby liquid control 'A' motor-driven pump in containment, Anticipated Transient Without Scram Control Panel in the control room, and the Division 1 'B' battery charger in the Division 1 DC (direct current) distribution room were free of potential adverse seismic conditions:

- anchorage was free of bent, broken, missing, or loose hardware;
- anchorage was free of corrosion that is more than mild surface oxidation;
- anchorage was free of visible cracks in the concrete near the anchors;
- anchorage configuration was consistent with plant documentation;
- SSCs will not be damaged from impact by nearby equipment or structures;
- overhead equipment, distribution systems, ceiling tiles, and lighting are secure and not likely to collapse onto the equipment;
- attached lines have adequate flexibility to avoid damage;
- the area appears to be free of potentially adverse seismic interactions that could cause flooding or spray in the area;
- the area appears to be free of potentially adverse seismic interactions that could cause a fire in the area; and
- the area appears to be free of potentially adverse seismic interactions associated with housekeeping practices, storage of portable equipment and temporary installations (e.g., scaffolding, lead shielding).

Observations made during the walkdowns that could not be determined to be acceptable were entered into the licensee's CAP for evaluation.

Additionally, inspectors verified that items that could allow the SFP to drain down rapidly were added to the Seismic Walkdown Equipment List and these items were walked down by the licensee.

b. Findings

No findings were identified.

4OA6 Management Meetings

.1 Exit Meeting Summary

On January 10, 2013, the inspectors presented the inspection results to the Site Vice President, Mr. Vito Kaminskas, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors verified that no proprietary information was retained by the inspectors or documented in this report.

.2 Interim Exit Meetings

- On October 5, 2012, the inspectors presented the inspection results of TI-2515/177 to Mr. V. Kaminskas and other members of the licensee's staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

- On November 29, 2012, the results of an Emergency Preparedness follow-up inspection for the RW building resin spill of June 3, 2012, were discussed with Mr. V. Kaminskas and other members of the licensee's management and staff. The inspectors confirmed that none of the potential report input discussed was proprietary.

- On November 30, 2012, the results of the ISFSI initial loading operational inspection were presented to Mr. V. Kaminskas and other members of the licensee's management and staff. Following additional information provided by the licensee, an additional exit meeting was held on December 14, with Mr. Zerr and other members of the licensee's staff. Licensee personnel acknowledged the information presented and the inspectors confirmed none of the potential report input discussed was considered proprietary.

- On December 7, 2012, the 2012 licensed operator requalification training biennial written examination and annual operating test technical debrief was presented to the licensed operator requalification lead instructor, Mr. R. Torres. The inspectors confirmed that none of the potential report input discussed was considered proprietary

- On December 14, 2012, the inspectors presented the inspection results for a radiological hazard assessment and exposure controls; RCS specific activity; occupational exposure control effectiveness; RETS/ODCM radiological effluent occurrences performance indicator verification; and TI-2515-185 inspections were discussed with Mr. V. Kaminskas and other members of the licensee's management and staff. The inspectors confirmed that none of the potential report input discussed was proprietary.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

V. Kaminskas, Site Vice-President
J. Grabnar, Site Operations Director
S. Baker, Radiation Protection Manager
H. Hanson, Performance Improvement Director
L. Lindrose, Security Manager
J. Oelbracht, Chemistry Manager
D. Reeves, Site Engineering Director
J. Tufts, Operations Manager
J. Veglia, Maintenance Director
T. Veitch, Director, Regulatory Compliance

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000440/2012005-01	NCV	Failure To Classify An Unusual Event (Section 1EP5.1)
05000440/2012005-02	NCV	Failure to Appropriately Control Access to a Locked High Radiation Area (Section 2RS1.1)
05000440/2012005-03	NCV	Inappropriate Procedures for Restoring LPCI Mode of RHR Following a LOCA at Mode 3 (Section 4OA5.1c.(1))
05000440/2012005-04	NCV	Deficiencies with Periodic Venting Procedures and Void Quantification (Section 4OA5.1c.(2))
05000440/2012005-05 07200069/2012002-01	NCV	Inadequate Fuel Handling Building Crane Maintenance Challenges Single-Failure-Proof Compliance (Section 4OA5.3b.(1))
05000440/2012005-06 07200069/2012002-02	SLIV	Failure to Follow Procedures That Ensure Safe Movement of a Dry Fuel Storage Canister (Section 4OA5.3b.(2))

Opened

05000440/2012005-07	URI	Follow-Up to Occupational Radiation Safety Performance Indicator Verification (Section 4OA1.2)

Closed

TI-2515/177	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (NRC Generic Letter 2008-01) (Section 4OA5.1)
TI-2515/158, Revision 1	Follow-Up On The Industry's Groundwater Protection Initiative (Section 4OA5.2)
TI-2515/188	Inspection of Near-Term Task Force Recommendation 2.3 Seismic Walkdowns (Section 4OA5.4)

LIST OF DOCUMENTS REVIEWED

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

<u>1R01 Adverse Weather Protection</u>
- PTI-GEN-P0026; Preparations for Winter Operation; Revision 6
- PTI-GEN-0027; Cold Weather Support System Startup; Revision 13
- NOP-WM-2001; Work Management Scheduling, Assessment and Seasonal Readiness Processes; Revision 14
- Licensee List of 2012 Winter Prep Orders; dated July 18, 2012
- Licensee List of 2012 Winter Prep Orders; dated October 24, 2012
- IOI-15; Seasonal Variations; Revision 19
- ONI-R36-2; Extreme Cold Weather; Revision 3

<u>1R04 Equipment Alignment</u>
- SOI-P45/P49; Emergency Service Water and Screen Wash System; Revision 20
- VLI-P45; Emergency Service Water System; Revision 12
- WO 200393675; 1B ESW Motor Power Cable Replacement
- WO 200393678; 1A ESW Motor Power Cable Replacement
- VLI-M43; Diesel Generator Building Ventilation System (Unit 1); Revision 4
- VLI-R44; Division 1 and 2 Diesel Generator Starting Air System (Unit 1); Revision 4
- VLI-R45; Division 1 and 2 Diesel Generator Fuel Oil System (Unit 1); Revision 5
- VLI-R46; Division 1 and 2 Diesel Generator Jacket Water Systems; Revision 4
- VLI-R47; Division 1 and 2 Diesel Generator Lube Oil; Revision 7
- VLI-R48; Division 1 and 2 Diesel Generator Exhaust, Intake and Crankcase Systems; Revision 6
- ELI-R71 (DG); Diesel Generator Building Lighting Panels; Revision 2
- SOI-R43; Division 1 and 2 Diesel Generator System; Revision 42
- SOI-R44; Division 1 and 2 Diesel Generator Starting Air System; Revision 14
- SOI-R45; Division 1 and 2 Diesel Generator Fuel Oil System; Revision 15
- SOI-R46; Division 1 and 2 Diesel Generator Jacket Water System; Revision 13
- SOI-R47; Division 1 and 2 Diesel Generator Lube Oil Systems; Revision 8
- SOI- M23/24; MCC, Switchgear, and Miscellaneous Electrical Equipment Area HVAC System; Revision 14
- VLI-M23/24; MCC, Switchgear and Miscellaneous Electrical Equipment Area HVAC System; Revision 7
- SOI-E51; Reactor Core Isolation Cooling System; Revision 30
- VLI-E51; Reactor Core Isolation Cooling System; Revision 8
- CR 2012-18493; Improper Fitting Make-Up On Flow Transmitter 0M23N0040A; dated November 27, 2012

<u>1R05 Fire Protection</u>
- FPI-0FH; Pre-Fire Plan Instruction – Fuel Handling Building; Revision 4
- FPI-1DG; Pre-Fire Plan Instruction – Diesel Generator Building; Revision 6
- FPI-0CC; Pre-Fire Plan Instruction – Control Complex; Revision 9
- FPI-SB; Pre-Fire Plan Instruction – Service Building; Revision 2
- FPI-0IB; Pre-Fire Plan Instruction – Intermediate Building; Revision 7

- ONI-P54; Fire; Revision 16

1R06 Flood Protection Measures
- Dwg 911-0617-00000; Auxiliary Building Drains; Revision G
- Dwg 911-0671-00000; Control Complex Drains; Revision F
- USAR Section 3.4.1; Flood Protection; Revision 14
- USAR Section 3.6; Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping; Revision 17
- Dwg 013-0005-00000; Plant Complex 620' Elevation and Above; Revision N
- Dwg 911-0671-00000; Control Complex Drains; Revision F
- Dwg 302-0741-00000; Liquid Rad Waste Sumps System; Revision W
- Maintenance Rule System Basis Document for System G60/G61, Miscellaneous and Liquid Radwaste Sumps; Revision 0
- WO 200297030; Lubricate Control Complex Laundry and Floor Drain Sump Pump 'B'; dated November 21, 2011
- WO 200270821; Lubricate Control Complex Laundry and Floor Drain Sump Pump 'A'; dated August 22, 2010
- WO 200036008; Cal Check Control Complex Floor Drain Sump Pump Control Switch; dated September 29, 2005

1R07 Annual Heat Sink Performance
- CR 2012-13700; PTI-P42-P0002, Emergency Closed Cooling 'B' Heat Exchanger Performance Test; dated September 6, 2012
- CR 2012-14817; Failure to Perform Heat Exchanger Testing As Required By Commitment L01916 to NRC Generic Letter (GL) 89-13; dated September 25, 2012
- WO 200342060; Eddy Current Inspection and Cleaning of PY-1P42B0001B Emergency Closed Cooling Heat Exchanger; dated May 6, 2011
- Calculation P42-33; Evaluation of the Heat Transfer Coefficient and Minimum Required Wall Thickness for Emergency Closed Cooloing Heat Exchangers 1P42-B0001A/B; Revision 1
- Calculation P42-051; Emergency Closed Cooling Heat Exchanger 'B' Loop Performance Test Evaluation; Revision 1
- DI-221; Perry Nuclear Power Plant ECC Heat Exchanger Design Input; dated May 2, 2012
- Test Protocol; First Energy Nuclear Operating Co., Perry Nuclear Power Plant, Emergency Closed Cooling Heat Exchangers; dated May 10, 2012

1R11 Licensed Operator Requalification Program
- NOP-OP-1002; Conduct of Operations; Revision 6
- Reactivity Plan – Perry Nuclear Power Plant; Evolution Specific – October 2012 Pattern Change; Revision 0; Update 0; dated October 26, 2012
- Procedure PYPB-PTS-005s Reviewed
- NOBP-NF-1013, Maintenance of The Training Simulator Core Model Fidelity, Revision 5
- NOP-OP-1013, Control of Time Critical Operator Actions, Revision 1
- PYBP-PTS-0033, Perry Cycle 14 BOL Core Test, Revision 8
- Perry Simulator Test Procedure for S3R MOC Cycle 14 Core
- Perry Licensed Operator Quarterly Watch Standing Log Records for 2011 and 2012 (various)
- 12-06-2012 Simulator Work Order Summary
- 12-06-2012 Open Simulator Work Orders
- 2011-2012 LORT Remediation Packages (various)
- NOP-TR-1001-01, Remedial / Makeup Recommendations (various), Revision 1
- NOP-TR-1001-02, FENOC Certifications (various), Revision 00

- NOP-TR-1001-03, FENOC De-Certifications (various), Revision 0
- 2012 Performance Improvement Plan (various crews and individuals)
- 2011 and 2012 License Renewal Records (various)
- PYPB-PTS-005, Operator Continuing Training Program Administration, Revision 36
- PYPB-PTS-0015, Job Performance Guide, Revision 5
- PYPB-PTS-0031, Simulator Review Board, Revision 5
- PYPB-PTS-0033, Simulator Configuration Control, Revision 10
- TMA-4206, Licensed Operator Requalification Programs, Revision 13
- PYPB-PTS-0033, 2012-00200 Simulator Physical Fidelity Evaluations, Attachment 5, Revision 10
- Six Licensed Operator Medical Records
- Simulator Performance Testing-Manual Scram
- Simulator Performance Testing-Trip of All Reactor Feedwater Pump Turbines
- Simulator Performance Testing-Main Steam Isolation Valve Closure
- Simulator Performance Testing-Simultaneous Trip of All Recirc Pumps
- Simulator Performance Testing-Trip of a Single Recirc Pump
- Simulator Performance Testing-Trip Main Turbine
- Simulator Performance Testing-Max Rate PWR Ramp Using Flow Control Valves
- Simulator Performance Testing-LOOP / LOCA Test
- Simulator Performance Testing-Main Steam Line Rupture Inside Drywell
- Simulator Performance Testing-Steady State Heat Balance Test 100% / 78% / 40%
- Plant Transient Comparison Tests-Closure of Turbine Control Valve #4
- Plant Transient Comparison Tests-Manual Reactor Scram Due To A Loss of Control Rod Drive Hydraulics
- Plant Transient Comparison Tests-Recirc Pump ‘A’ Trip
- CR 2012-03044, Faster Than Expected Air Header Pressure Drop During Valve Testing; dated February 27, 2012
- CR 2011-96712, Post Event Debrief PYBP-POS-2-1 For #4 CV Closure; dated June 21, 2011
- Perry Evaluation Scenarios:
  - PC5B Revision 0, Week 7;
  - RP1C Revision 0, Week 7;
  - RP2D Revision 0, Week 7;
  - PC1D Revision 0, Week 3;
  - RP1A Revision 0, Week 3;
  - RP2E Revision 0, Week 3.
- Job Performance Measures:
  - JPM E12-012;
  - JPM C61-002;
  - JPM C11-513;
  - JPM C51-501;
  - JPM D51-301;
  - JPM P50-001;
  - JPM P52-501;
  - JPM E12-014;
  - JPM P43-501;
  - JPM C41-001;
  - JPM C11-014;
  - ADM-306, SRO only

- Biennial Written Examinations:
  - Week 6, 2012 Biennial Written Exam for Reactor Operators;
  - Week 6, 2012 Biennial Written Exam for Senior Reactor Operators;
  - Week 7, 2012 Biennial Written Exam for Reactor Operators; and
  - Week 7, 2012 Biennial Written Exam for Senior Reactor Operators.

<u>1R12 Maintenance Effectiveness</u>

- WO 200505382; Troubleshoot Flow Control Valve 'B' Oscillations; dated June 21, 2012
- Notification 600667737; Develop a Business Case for Jet Pump Cleaning; dated February 23, 2012
- CR 2011-96328; Annunciator Recirc B Outer Seal Leakage Is Locked In; dated June 12, 2011
- CR 2012-10964; Oil Samples for B33, C85 and N32 Have Not Been Sent To The Lab; dated July 12, 2012
- CR 2012-09427; Flow Control Valve 'B' Oscillations; dated June 7, 2012
- CR 2011-89545; Maximum Allowed Core Flow Not Obtainable; dated February 14, 2011
- SVI-C71-T5233; RPS – Electrical Power Monitoring Calibration/Functional; Revision 11
- Plant Health Committee Presentation On C-71 Reactor Protection System; Delivered on November 19, 2012
- CR 2012-15887; System Ranking for RPS Is Red for 3rd Quarter 2012; dated November 9, 2012
- CR 2011-01862; Anomaly During Performance of SVI-C71-T0046, Section 5.1.4; dated September 13, 2011
- CR 2012-11466; Loss of RPS 'A', ONI-C71-2 Entry; dated July 23, 2012
- WO 200201338; “New PM” Overhaul Division 'A' RPS MG Set; dated September 14, 2010
- WO 200514209; Replace Voltage Regulator On RPS 'A' MG Set; dated August 2, 2012
- CR 2012-11628; Delay In Returning RPS MG 'A' to Service Due to Replacement Part Lacking Required 100 Hour Burn-In; dated July 26, 2012
- Perry Power Station Protected Equipment Addendum to Support Division 1 ECCS Outage; dated October 4, 2012
- CR 2012-11511; Potential MSIV Isolation and Reactor SCRAM Near Miss Due To Failed RPS MG 'A'; dated July 24, 2012
- CR 2012-19606; Work Not Released As Scheduled in PWIS; dated December 18, 2012
- Component Equipment Replacement Package #687; Revision 0
- CR 2012-02077; Documentation of As-Found Condition On Division 2 EDG Turbo Intercooler Support Gusset Plate; dated February 8, 2012
- CR 2012-18961; NDE Exam Results of Division 2 EDG Intercooler Crack Require Evaluation; dated December 5, 2012
- CR 2010-78036; Failure of Division 2 EDG Right and Left Bank Starting Air Systems; dated June 9, 2010
- CR 2010-71395; Repetitive Human Error Trap Needs Eliminated; dated February 10, 2010
- ECP #09-0638-001; Revise Reset Values for R44 Starting Air System Low Pressure Alarm Pressure Switches; Revision 0
- ECP #09-0596-002; Replace Barksdale Pressure Switch Model E15-H500 for R44 Standby Diesel Generator Starting Air System Pressure Switches; Revision 0
- WO 200414138; “IPO-36” Division 2 EDG Starting Air Pressure; Revision 0
- WO 200385940; Replace Pressure Switch Under ECP 09-0956-001; dated July 15, 2010
- WO 200385941; Replace Pressure Switch Under ECP 09-0956-002; dated October 18, 2010
- CR 2012-17921; Order Package Submitted to QC for Closure Rejected; dated November 12, 2012

1R13 Maintenance Risk Assessments and Emergent Work Control
- Locked High Radiation Area Controls for Dry Cask Storage; Revision 0
- WO 200410106; PY-1G58D0505 Cask Load # 3 (First Campaign) Hi-Storm # 351; dated November 13, 2012
- ALARA Plan 120156; Campaign # 3 1G58D0505 Load Spent Fuel Into Hi-Storm # 351 Transport to Storage and Support Work; dated October 26, 2012
- NOBP-OP-0007-01; IPTE Worksheet; Revision 1 – Titled: Dry Fuel Storage Campaign – 2012 Campaign 1, Revision 1; dated November 5, 2012
- eSOMS Narrative Logs dated October 29, 2012
- NOP-OP-1007; Risk Management; Revision 15

1R15 Operability Determinations and Functionality Assessments
- EER Notification 600797776; Torque Criteria Verification; dated November 15, 2012
- WO 200438889; Inspect 50 percent of Cams/Valves/Tap of Division 2 EDG; dated November 12, 2012
- CR 2012-18171; Lockwasher Missing On #6 Right Bank Injector Control Arm: dated November 16, 2012
- CR 2012-18173; Delay In Division 2 Operability Due To Missing Lockwasher; dated November 16, 2012
- CR 2012-18110; HU Error: Different Torque Values Discovered During Re-Performance of PMI-0010 Per Work Order 200438889; dated November 15, 2012
- CR 2012-18078; Weld Broke On Bracket Supporting Fuel Lines; dated November 15, 2012
- CR 2012-18457; 1P52-F160 Failed Stroke Closed Time; dated November 26, 2012
- WO 200538884; Valve Failed Stroke Time / CR 2012-18457; dated November 30, 2012
- Notification 600799170; Valve 1P52-F160, Personnel Airlock Elevation 603' Supply Air Outboard Isolation Valve, Failed Its Stroke Time During SVI-P52-T2002; dated November 26, 2012
- CR G202-2009-61450; Reactor Recirc FCV B Drifts In Closed Direction When Locked Up; dated July 6, 2009
- eSOMS Narrative Logs dated November 14-15, 2012
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated March 21, 2001
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated May 9, 2003
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated March 30, 2005
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated April 12, 2005
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated May 3, 2007
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated May 9, 2007
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated June 24, 2007
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated April 29, 2009
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated May 26, 2011
- SVI-B33-T1158; Reactor Recirculation Flow Control Valve Functional Test; dated May 31, 2011
- PNPP Potential Issue Form; PIF No. 97-0622; dated April 9, 1997
- CR 2012-19063; Degraded Shading Coil Found In Starter for EF1C12-J; dated December 6, 2012
- CR 2012-19084; Degraded Over Load Relay Found In EF1C12-J; dated December 7, 2012

1R18 Plant Modifications

- ECP 12-0056; Rewire Control Room Ammeter (1P45-R010) To Prevent A Hot Short In The Ammeter Circuit From Tripping Switchgear EH1106; Revision 2
- ECP 11-0422-001; Temporary Separation Of Control Room Ammeter Circuit For Breaker EH1106; Revision 1
- ECP 11-0422-003; Temporary Mod To Remove TM (Jumper) For Control Room Ammeter 1P45- R010; Revision 3
- WO 200468289; Implement Temp Mod 11-0422-001 To Bypass Ammeter For EH1106 In The Control Room; dated July 8, 2011
- WO 200468430; Implement ECP 12-0056-002 For ESW Pump 'A', 1P45-C001A and Breaker EH1006; dated October 4, 2012
- WO 200476106; On-line Noble Chemistry – Install Temporary Modification; dated February 16, 2012
- ECP 08-0183-007; Temp Mod for ONLC application; Revision 1

1R19 Post-Maintenance Testing

- WO 200447062; PY-1E12 Residual Heat Removal; dated October 3, 2012
- SVI-E12-2001; RHR 'A' Pump and Valve Operability Test; Revision 30
- WO 200468430; PY-1P45C0001A ESW Pump 'A'; dated October 5, 2012
- SOI-P45/P49; Emergency Service Water and Screen Wash Systems; Revision 20
- WO 200438900; ESW Pump 'B' Breaker Removal From Pocket To Install and Remove Ground Truck; dated November 6, 2012
- SVI-P45-T2002; ESW Pump 'B' and Valve Operability Test; Revision 29
- SVI-R43-T1318; Diesel Generator Start and Load Division 2; Revision 14
- SOI- R43; Division 1 and 2 Diesel Generator System; Revision 42
- SVI-E12-T2002; RHR ‘B’ Pump and Valve Operability Test; Revision 30

1R22 Surveillance Testing

- SVI-C41-T2001-A; Standby Liquid Control 'A' Pump and Valve Operability Test; Revision 17; dated October 1, 2012
- SVI-E51-T2001-1; RCIC Pump and Valve Operability Test; Revision 36; dated October 31, 2012
- WO 200511822; RCIC Steam Supply First Drain Shutoff; dated October 31, 2012
- SVI-M15-T1239B; Annulus Exhaust Gas Treatment Train 'B' Operability Test; Revision 0; dated November 20, 2012
- PRI-TSR; Technical Specification Rounds; Revision 29
- OAI-1702; Operations Section Rounds Sheets, Logs and Records; Revision 12
- SVI-E31-T0374; Reactor Cooling System Unidentified Leakage Determination; Revision 4

1EP4 Emergency Action Level and Emergency Plan Changes

- Perry Nuclear Power Plant Emergency Plan; Revisions 33, 34, 35, 36, and 37
- EPI-A-0001; Emergency Action Levels; Revision 24

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

- PNPP Emergency Plan; Sections 3.0 and 4.1; Revision 35
- EPI-A1; Emergency Action Levels; Revision 24
- HPI-D0001; Radiation and Contamination Survey Techniques; Revision 22
- NORM-OP-4000; Radiation Protection Definitions and References; Revision 1
- ONI-D17; High Radiation Levels With Plant; Revision 16
- PSI-0019; Emergency Action Level (EAL) Bases Document; Revision 15

- Radwaste 574' Elevation East and West Area Radiation Monitor Chart Printouts; dated June 2 – 8, 2012
- CR 2012-18622; Unusual Event Notification Completed for June 3, 2012, Resin Spill; dated November 29, 2012
- CR 2012-13039, ERO Drill Scenario Did Not Identify Unusual Event Criterion; dated August 24, 2012
- CR 2012-09729; Unusual Event Not Declared; dated June 14, 2012
- CR 2012-09447; Radwaste 574' Elevation Survey; dated June 7, 2012

<u>2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)</u>
- CR 2011-98372; Material Handling; dated July 28, 2011
- CR 2012-07583; Challenges Identified for Inadvertent Climbing and Access To Locked High Radiation Areas; dated August 10, 2012
- CR 2012-07595; Unsafe Scaffolds Found Green Tagged During Catacomb Walkdown; dated May 9, 2012
- CR 2012-09447; Pre-Job Survey of Rad Waste 574' Elevation; dated June 6, 2012
- CR 2012-11096; During Routine Survey Increased Dose Rates Found in Area; dated July 16, 2012
- CR 2012-15636; Spill Caused by Hose Connection Disconnected During the Blow Down Process on Dry Cask HI-TRAC/MPC #2; dated October 5, 2012
- CR 2012-15719; Water On Floor Rad Waste 646' By Waste Collector Filtrate Tank; dated October 6, 2012
- CR 2012-15753; Filtrate Tank Controller Degrader and May Be Spreading Contamination; dated October 7, 2012
- CR 2012-15942; Possible Procedure Issue During Draining of Annulus for Dry Cask Storage; dated October 9, 2012
- CR 2012-15953; Water On Floor of The Waste Collector Filter Room; dated October 9, 2012
- CR 2012-15964; Ladders Staged Next to LHRA; dated October 10, 2012
- CR 2012-16048; Inadvertent Access to LHRA Due To Scaffold Near Labyrinth; dated October 11, 2012
- NOP-OP-4101; Access Controls for Radiologically Controlled Areas; Revision 08
- NOP-OP-4102; Radiological Postings, Labeling, and Markings; Revision 09
- RWP 120134; HCA Strongback Cutting and Removal; dated May 2012
- RWP 120141; C510001E TIP Drive Activities; dated June 2012
- RWP 120206; 0G50D0001 Waste Collector Filter Room Belt Replacement – HRA/Airborne; dated October 2012

<u>4OA1 Performance Indicators (71151)</u>
- NOBP-LP-4012; NRC Performance Indicators; Revision 04
- NOBP-LP-4012-09; NRC Performance Indicator Data Sheets; Barrier Integrity Reactor Coolant System Dose Equivalent Iodine; July 2011 through September 2012
- NOBP-LP-4012-14; NRC Performance Indicator Data Sheets; Occupational Radiation Safety; July 2011 through September 2012
- NOBP-LP-4012-15; NRC Performance Indicator Data Sheets; Public Radiation Safety; April 2011 through September 2012
- NOBP-LP-4012-10; NRC Performance Indicator Data Sheets; Reactor Coolant System Leakage October 2011 through September 2012

4OA2 Identification and Resolution of Problems (71152)

- CR 2012-18840; PM Order # 2004-77803 Past Its Overdue Date; dated December 3, 2012
- CR 2012-13561; Thermal Overloads Tripped On Manhole # 6 Under Drain Pump; dated September 4, 2012
- CR 2012-12172; Under Drain Manhole # 7 Pump PY-OP72C0001D Is Running Continuously; dated August 7, 2012
- CR 2012-08881; Inability to Close PFA Associated With The Plant Under Drain System; dated May 29, 2012
- CR 2011-07169; PTI-P72-P0005 Plant Underdrain Groundwater Level Readings Non-conservative Acceptance Criteria; dated December 21, 2011
- CR 2012-06950; MS-ID: In Leakage Observed on AUX 574'; dated December 16, 2011
- NOP-LP-4008; Licensing Document Change Process; Revision 3
- PYBP-SITE-0039; PNPP USAR Validation Access Database; Revision 2
- PTI-P72-P0005; Plant Underdrain Groundwater Level Readings; Revision 6
- Dwg D-302-861; Plant Foundation Underdrain System; Revision 14
- Calculation 3.1.3.1; G50 Declassification Study; Revision 9
- Calculation 3.1.3.2; G50 Declassification Study – Dose; Revision 8
- Aggregate Risk To Perry Station Operations Matrix; Third Quarter 2012
- CR 2012-14283; RHR B Minimum Flow Valve Lost Indication When Opened From the Control Room; dated September 18, 2012
- CR 2012-18333; Manhole Pump #6 Running Intermittently; dated November 20,2012
- Operator Burden List dated November 19, 2012
- Operations Decision Making Issue for NCC Leakage; dated February 1, 2012
- NOP-OP-1010; Operational Decision-Making; Revision 4
- CR 2012-18466; System Components Degradation Leads to Operator Work-Arounds to Operate System; dated November 26, 2012

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

- CR 2012-18576; As-Found Condition of Acetylene Bottle Found In Radwaste Control Room; November 28, 2012
- CR 2012-18589; Entered Unusual Event (MU1) for Toxic Gas (Carbon Monoxide) In The Radwaste Control Room; dated November 28, 2012
- CR 2012-18600; Work Delays In Electrical Maintenance Caused By Unusual Event Could Drive the Work Window Into Orange Risk; dated November 28, 2012
- CR 2012-18604; Gas Cylinder Was Taken Into a Confined Space; dated November 28, 2012
- CR 2012-18608; Workers Receive An Alarm for Hazardous Atmosphere; dated November 28, 2012
- CR 2012-18609; NOBP-TR-1122 Post Event Critique Results for Fire Brigade Response to Unusual Event; dated November 28, 2012

4OA5 Other Activities

- NEI APC 09-20; GL 2008-01, Evaluation of Unexpected Voids or Gas Identified In Plant ECCS and Other Systems; dated May 18, 2008
- NEI 09-10; Guidelines for Effective Prevention and Management of System Gas Accumulation; Revision 0
- SOI-E12; Residual Heat Removal System; Revision 57
- SVI-E22-T1183; HPCS Valve Lineup Verification and System Venting; Revision 11
- SVI-E21-T1181; LPCS Venting and Valve Lineup Verification; Revision 13
- SVI-E12-T1182-C; RHR C LPCI Valve Lineup Verification and System Venting, Revision 14
- SVI-E12-T1182-B; RHR B LPCI Valve Lineup Verification and System Venting, Revision 10

- SVI-E12-T1182-A; RHR A LPCI Valve Lineup Verification and System Venting, Revision 12
- ARI-H13-P601-0021-F6; LPCS Pump Discharge Pressure High; Revision 15
- ONI-SPI A-9; RHR A SDC Fill and Vent; Revision 0
- NOP-CC-5712; Ultrasonic Detection of Gas Voids in Liquid Systems Using the EPOCH LTC; Revision 0
- BOP-UT-12-007; UT Report for SVI-E21-1181 Piping Near Flow Element 1E21N0656; dated July 5, 2012
- BOP-UT-12-004; UT Report for SVI-E12-1182C Piping Below Valve 1E12F0067; dated April 13, 2012
- BOP-UT-12-003; UT Report for SVI-E21-1181 Piping Near Flow Element 1E21N0656; dated April 4, 2012
- NOP-OP-1009; Operability Determinations and Functionality Assessments; Revision 3
- NOP-LP-2001; Corrective Actions Program; Revision 30
- Notification 600660960; PNPP UFSAR Changes for Gas Management; dated January 10, 2011
- GEN-019; ECCS Void Acceptance Criteria; Revision 0
- GEN-019; Addendum to ECCS Void Acceptance Criteria; Revision 0
- E22-C02; HPCS – CST Low Level Transfer Trip – 1E22-N654C(G); Revision 6
- P11-012; The Condensate Storage Tank (CST) Vortexing Analysis; Revision 2
- E22-043; HPCS System Hydraulic Analysis; Revision 0
- E12-088; RHR System Hydraulic Calculation; Revision 1
- 304-0644-00107; RHR Piping Isometric Sheet 2; Revision E
- 304-0701-00101; HPCS Piping Isometric Sheet 2; Revision B
- 304-0641-00114; RHR-Auxiliary Building Piping Isometric Sheet 2; Revision B
- 302-0701-00000; HPCS P&ID; Revision JJ
- 302-0624-00000; RHR P&ID; Revision HH
- 302-0705-00000; LPCS P&ID; Revision FF
- WO 200421560; OE Evaluation of NRC IN 2010-11; undated
- WO 200508067; HPCS Valve Lineup Verification and System Venting; dated September 15, 2012
- WO 200489207; LPCS Venting and Valve Lineup Verification; dated July 3, 2012
- WO 200489122; RHR 'A' LPCI Valve Lineup and System Venting; dated July 4, 2012
- WO 200456303; RHR 'B' LPCI Valve Lineup and System Venting; dated September 9, 2012
- CR 2011-96588; SVI-E22-T1183 Air Found At Vent Point Greater Than 1 Second; dated June 17, 2001
- CR 2011-90863; Air Observed During RHR 'C' LPCI SVI; dated March 11, 2011
- CR 2009-59349; Air Noted During LPCS Venting and Valve Lineup Verification SVI-E21-T1181; dated May 17, 2009
- CR 2012-14134; Review of NRC TI-177 Inspection Has Identified the Potential for Pre-Conditioning of Venting Procedures; dated September 14, 2012
- CR 2012-14244; NRC TI-177 ECCS Suction and Discharge Piping Void Acceptance Criteria; dated September 17, 2012
- CR G202-2009-57951; Excessive Venting From RHR 'B' HX Vent Valve; dated April 25, 2009
- CR G202-2011-95230; SVI-E12-T1182-B Air Found During High Point; dated May 20, 2011
- CR G202-2009-55101; Possible Air in System; dated March 3, 2009
- CR G202-2009-55054; Air Observed During RHR 'B' High Point Fill and Vent; dated March 10, 2009
- CR 2012-15564; NRC TI-177 Inspection - Detail Evaluation of Air Slug Transport to ECCS Pump Impeller Was Not Provided In Revision 1 To Calculation GEN-019; dated October 4, 2012

- CR 2012-15554; NRC TI-177 Inspection - Calculation P11-012, Revision 2 Deficiency; dated October 4, 2012
- CR 2012-15550; NRC TI-177 Inspection - In Mode 3, With Operation of An RHR Train Is SDC Mode, If a LOCA Occurs, Recovery of That RHR Train Does Not Appear To Be Supported By Current Procedures; dated October 4, 2012
- CR 2012-15065; NRC TI-177 Inspection - NRC Inspector Concern That Dynamic Venting of ECCS Systems May Need To Be Performed On A Frequent Basis; dated September 27, 2012
- CR 2012-14947; NRC TI-177 Inspection - Investigation Timeliness for Condition Reports Identifying ECCS Piping Air Voids; dated September 26, 2012
- CR 2012-14830; NRC TI-177 Inspection – ONI Procedure To Fill and Vent Shutdown Cooling Piping Does Not Fully Address High Temperature Water Considerations; dated September 25, 2012
- CR 2012-14509; NRC TI-177 Inspection – Generic Letter 2008-01 Response for High-Low Pressure Interface Evaluation Lack Sufficient Detail/Rigor; dated September 20, 2012
- CR 2012-14487; NRC TI-177 Inspection – ECCS Venting Procedures May Need Clarification; dated September 20, 2012
- CR 2012-14380; NRC TI-177 Inspection – Calculation GEN-019 Does Not Provide Sufficient Detail/Rigor for Determining Void Size; dated September 19, 2012
- CR 2012-14269; TI-177 A-01 to Calculation GEN-019, Revision 0 Contains Editorial Error; dated September 17, 2012
- CR 2012-14311; TI-177, NRC Identified GEN-019, A-01 Does Not Meet All APC 09-20 Requirements; dated September 18, 2012
- Operations Standing Order – RHR Shutdown Cooling Operation in Mode 3; dated October 4, 2012
- FM-067; Assemblies and Decay Heat for First Spent Fuel Dry Storage Campaign; Revision 0
- FTI-D0002; Special Nuclear Material Physical Inventory; Revision 9
- Quality Assurance Program Manual; dated April 26, 2010
- NOP-OP-4005; ALARA Program; Revision 1
- ALARA Plan 120154; Revision 0
- G-58-P-006; Fire Hazards Analysis for the Dry Cask Storage System Inside the Fuel Handling Building; Revision 0
- G58-S-R-L-006; NUREG-0554 Conformance Matrix for Fuel Handling Area Crane; Revision 0
- G58-S-R-U-012; ISFSI Radiological Monitoring Requirements; Revision 0
- G58-H-HI-2083959; Consequences of HI-STORM Blocked Duct Accident Condition; Revision 0
- HPI-D0005; RP Monitoring Requirements for Dry Fuel Storage Loading Operations; Revision 4
- HPI-D0006; Independent Spent Fuel Storage Installation Radiation Survey; Revision 0
- PI-901247-01; Closure Welding of Multi-Purpose Canisters at Perry 1 (GPRY) Nuclear Station; Revision 0
- GQP-9.2; High Temperature Liquid Penetrant Examination and Acceptance Standards for Welds, Base Materials and Cladding (50 °F - 300 °F); Revision 4
- GQP-9.6; Visual Examination of Welds; Revision 10
- WCP-5; Weld and Base Metal Repair; Revision 5
- 8 MC-GTAW; Welding Procedure Specification; Revision 12
- 8-NG MC-GTAW; Welding Procedure Qualification Record; Revision 0
- 8 MN-GTAW; Welding Procedure Specification; Revision 0
- MSLT-DSC-PCI; Helium Mass Spectrometer Leak Test Procedure; Revision 0
- IOI-19; Spent Fuel Dry Cask Storage; Revision 0
- PAP-1313; Control of Lifting Operations; Revision 15
- PMI-0031; Fuel Handling Area Crane Preventative Maintenance; Revision 7

- Perry Nuclear Power Plant Independent Spent Fuel Storage Installation (ISFSI) 10 CFR 72.212 Evaluations Report; Revision 0
- G58-S-R-L-011; Evaluation of Fire and Explosion Hazards for ISFSI; Revision 0
- G58-S-R-M-004; Evaluation of Hydrogen and Oxygen Hazards for ISFSI; Revision 0
- PAP-1910; Fire Protection Program; Revision 27
- GMI-0210; MPC Loading; Revision 6
- GMI-0212; Vacuum Drying System Operations; Revision 0
- GMI-0213; MPC Sealing; Revision 3
- GMI-0215; Response to Abnormal Condition; Revision 2
- GMI-0221; HI-STORM Movement With Stack-Up Seismic Restraint Installed; Revision 5
- GMI-0222; MPC Transfer With Stack-Up Seismic Restraint Installed; Revision 6
- SFDS Project Readiness for Second Canister Loading; dated September 30, 2012
- WO 200410103; Trouble Shoot Port Drain Cap Leak; dated September 19, 2012
- WO 200465721; Periodic Maintenance on FHB Crane; dated January 30, 2012
- WO 200519103; Support Removal of HI TRAC From Cask Pit; dated August 30, 2012
- Kone Cranes Memo; Perry Fuel Handling Building Crane, P&H #CN-25590 Operation After Magnetorque Checks; dated August 31, 2012
- Maintenance Manual; EPD-100 Static Stepless Control
- CR 2012-12748; Concrete Spall and Crack At Face of IB8 Wall Under FHB Roof Girder; dated August 17, 2012
- CR 2012-12933; Dry Cask: FHB Crane Reported To Have Sparks During Load Movement; dated August 22, 2012
- CR 2012-13005; Procedure GMI-0071 Needs Revised For Fuel Handling Building Crane; dated August 24, 2012
- CR 2012-13085; FHB Crane 0L51E0003 Magnetorque Scrubbing Between The Stationary and Rotating Members; dated August 25, 2012
- CR 2012-13086; FHB Crane 0L51E0003 Magnetorque Inspection; dated August 25, 2012
- CR 2012-13003; Deficiencies in PMI-0031 “Fuel Handling Are Crane Preventative Maintenance” Section 5.8 “Main Hoist Magnetorque” Maintenance; dated August 24, 2012
- CR 2012-13109; Fuel Handling Building Crane Continuously Tripping After 15 Minutes During Operation; dated August 27, 2012
- CR 2012-13110; Grinding Noise Coming From Fuel Handling Building Crane Magnetorque; dated August 27, 2012
- CR 2012-13234; Abnormal Amp Readings Identified On Magnetorque Circuitry; dated August 28, 2012
- CR 2012-13240; Fuel Handling Building Crane PM Not Performed As Recommended Per The Vendor Manual; dated August 29, 2012
- CR 2012-13315; NRC Question Associated With FHB Crane Being Single Failure Proof Without Magnetorque; dated August 29, 2012
- CR 2012-13484; PY-PA-12-02 FHB Crane (OL51E0003) Magnetorque Preventive Maintenance Gap; dated August 31, 2012
- CR 2012-13930; Fuel Handling Bridge Main Grapple Clearance Issue When Loading Multi-Purpose Canister; dated September 11, 2012
- CR 2012-13967; SFDS Procedure GMI-0213 Contains Two (2) Typos in Section 5.2 That Was Identified by the SFDS Inspector; dated September 12, 2012
- CR 2012-14035; GMI-0213 – “MPC Sealing,” Hydrostatic Test Rig For MPC Has No Backflow Preventer for Cross Contamination Control. SFDS NRC Inspector Identified; dated September 13, 2012
- CR 2012-14033; Interpretation Issue With Regard To Procedure Adherence vs Order Adherence; dated September 13, 2012

- CR 2012-14025; Fatigue Management Issue; dated September 13, 2012
- CR 2012-14063; NRC Question: Evaluation of Fire Hazards For DCSS Inside the FHB; dated September 13, 2012
- CR 2012-14149; FME Found In Cask Pit Pool Shelf; dated September 15, 2012
- CR 2012-14169; Work Activity Conducted Out of Schedule Sequence; dated September 15, 2012
- CR 2012-14170; Enhancement Actions Identified by The NRC for Decon of HI-TRAC; dated September 15, 2012
- CR 2012-14184; Issues Identified With Dry Cask Storage Hot Work Permit; dated September 15, 2012
- CR 2012-14361; Unexpected Helium Leak While Performing MPC Sealing for The Dry Fuel Storage Project; dated September 19, 2012
- CR 2012-14613; Missed Order Operations and Notification While Performing Work 200410107 for Dry Cask Storage; dated September 22, 2012
- CR 2012-14468; Improvement Opportunity To Better Incorporate RP Into Emergent DCS Work; dated September 20, 2012
- CR 2012-14634; Locked High Radiation Area HI-STORM/MPC Access; dated September 23, 2012
- CR 2012-14878; Procedural Guidance In Use of Infrared Pyrometer Is Lacking In GMI-0222 "MPC Transfer With Stack-Up Seismic Restrain Installed;" dated September 25, 2012
- CR 2012-14977; Mating Device Drawer Was Closed for 4.5 Hours While Loaded MPC Was Inside HI-STORM. HI-STORM was Positioned In The ZPT; dated September 26, 2012
- CR 2012-18567; Dry Cask: No Written Basis for MPC Cooling With 24" Mating Device Drawer Opening (GMI-022) NRC Identified; dated November 28, 2012
- CR 2012-18602; NRC Question As To Why MPC Lid Welding Indication Was Not Considered A "Weld Repair," dated November 28, 2012
- CR 2012-19263; NRC Question On Monitoring "Section Thickness" During Dry Cask Storage Welding; dated December 11, 2012

<u>4OA5 Other Activities (Temporary Instruction 2515/185 Revision 1)</u>

- CR 2010-72723; No Long Term Plan Has Been Established for Preventive Maintenance; dated March 5, 2012
- FirstEnergy Groundwater Monitoring Well Installation and Monitoring Report; Perry Nuclear Power Station; dated June 27, 2008
- GAT 600601049; Tritium Recapture Study; dated March 1, 2010
- NEI 07-07; Industry Groundwater Protection Initiative – Final Guidance Document; dated August 2007
- NOP-OP-4705; Response to Contaminated Spills/Leaks; Revision 06
- NOP-OP-2012; Ground Water Monitoring; Revision 06
- NOP-ER-2007; Buried Pipe Integrity Program; Revision 04
- NOP-WM-4007; Excavation and Trenching Controls; Revision 02
- PAP-0809; Radiological Environmental Contamination Response; Revision 3
- Perry Operations Manual - Offsite Dose Calculation Manual; Revision 19
- Perry Nuclear Power Plant Annual Environmental and Effluent Release Reports for 2010 and 2011; dated March 2011, April 2012, respectively
- Perry Nuclear Power Plant Ground Water Field Sampling Plan; dated August 3, 2007
- Perry Nuclear Power Plant Ground Water Flow Characteristics Report; dated October 20, 2006
- SN-SA-2012-0262; RETs/REMO Program Health Self-Assessment; dated December 2012

4OA5 Other Activities (Temporary Instruction 2515/188)

- Perry NPP Near-Term Task Force Recommendation 2.3 Seismic Walkdown Report; dated September 28, 2012
- Seismic Walkdown Checklist Items for:
  1E22N0005; Flow Transmitter
  1E22F0004; HPCS Injection Valve
  1E22C0001; HPCS Pump
- Seismic Walkdown Equipment Lists Category 1 and 2
- CR 2012-12331; PNPP IPEEE Identified Vulnerabilities Not Adequately Addressed; dated August 9, 2012
- CR 2012-12373; Possible Undocumented Modification Installed in the Plant; dated August 10, 2012
- CR 2012-12335; Control Room Ceiling Tiles Not Meeting Industry Good Practice/Standards for Seismic Considerations; dated August 9, 2012
- CR 2012-12375; Potential Seismic II/I Concern Identified in the Plant; dated August 10, 2012
- CR 2012-12222; Unrestrained Operations Electrical Locker (Tool Box Panel) Near Panel 1R22-S0006 Containing Bus EH12; August 7, 2012

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
ALARA	As-Low-As-Is-Reasonably-Achievable
CAP	Corrective Action Program
CBST	Condensate Backwash Settling Tank
CDF	Core Damage Frequency
CR	Condition Report
CFR	*Code of Federal Regulations*
DNMS	Division of Nuclear Materials Safety
EAL	Emergency Action Level
ECCS	Emergency Core Cooling System
ECP	Engineering Change Package
EDG	Emergency Diesel Generator
ESW	Emergency Service Water
°F	Degree Fahrensheit
FHB	Fuel Handling Building
FSAR	Final Safety Analysis Report
GL	Generic Letter
GMI	General Maintenance Instruction
GPI	Groundwater Protection Initiative
HI-STORM	Storage Cask
HI-TRAC	Transfer Cask
HPCS	High-Pressure Core Spray
IMC	Inspection Manual Chapter
IN	Information Notice
IP	Inspection Procedure
IR	Inspection Report
ISFSI	Independent Spent Fuel Storage Installation
IST	Inservice Testing
JPM	Job Protective Measures
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
LORT	Licensed Operator Requalification Testing
LPCI	Low-Pressure Coolant Injection
LPCS	Low-Pressure Core Spray
MCID	Materials Control ISFSI and Decommissioning
MPC	Multi-Purpose Canister
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
OCC	Outage Control Center
ONI	Off-Normal Instruction
OWA	Operator Workaround
PI	Performance Indicator
PMI	Preventive Maintenance Instruction
PNPP	Perry Nuclear Power Plant
RCS	Reactor Coolant System
RW	Radwaste (Radioactive Waste)

RHR	Residual Heat Removal
RP	Radiation Proctection
SAT	Systems Approach to Training
SDC	Shutdown Cooling
SDP	Significance Determination Process
SL	Severity Level
SFP	Spent Fuel Pool
SRO	Senior Reactor Operator
SVI	Surveillance Instruction
SSC	Structure, System, and Component
TI	Temporary Instruction
TS	Technical Specification
URI	Unresolved Item
USAR	Updated Safety Analysis Report
ZPT	Zero Profile Transporter

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

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

Sincerely,

/RA/

Michael A. Kunowski, Chief
Branch 5
Division of Reactor Projects

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