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

November 4, 2009

Mr. Mark Bezilla
Site Vice President
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 05000440/2009004

Dear Mr. Bezilla:

On September 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on October 7, 2009, with you and members of your staff.

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

Based on the results of this inspection, four self-revealed findings of very low safety significance (Green) were identified. Three of the findings were determined to involve violations of NRC requirements. Additionally, two licensee-identified violations, which were determined to be of very low safety significance, are listed in Section 4OA7 of this report. However, because of the findings' very low safety significance and because they are entered into your corrective action program, the NRC is treating the findings as non-cited violations consistent with Section VI.A.1 of the NRC Enforcement Policy.

If you contest any NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, Region III, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Perry. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at Perry. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

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

Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

Docket No. 50-440
License No. NPF-58

Enclosure: Inspection Report 05000440/2009004
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServ

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	050000440/2009004
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, Ohio
Dates:	July 1, 2009, through September 30, 2009
Inspectors:	M. Marshfield, Senior Resident Inspector J. Bashore, Acting Senior Resident Inspector M. Munir, Reactor Inspector R. Murray, Reactor Engineer M. Phalen, Health Physicist P. Voss, Reactor Engineer G. Wright, Project Engineer
Approved by:	Jamnes L. Cameron, Chief Branch 6 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000440/2009004; 07/01/2009 – 09/30/2009; Plant Modifications; Other Activities.

The inspection was conducted by resident and regional inspectors. The report covers a 3-month period of resident inspection. Four green findings, three of which were non-cited violations (NCVs) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609 "Significance Determination Process" (SDP). Cross-cutting aspects were determined using IMC 0305, "Operating Reactor Assessment Program." Findings for which the SDP does not apply may be "Green," or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Event

- Green. 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," was self-revealed for the licensee's failure to have an appropriate troubleshooting plan for repairing Average Power Range Monitor (APRM) 'A.' Specifically, the troubleshooting plan for inoperable APRM 'A' did not provide proper guidance to the technicians resulting in an unexpected half scram on the reactor protection system and subsequent required operator actions. The licensee entered the error into their corrective action program as CR 09-63991. As part of its corrective actions, the licensee planned to place placards in the APRM cabinets warning of the special instructions to remove and replace the cards.

  The finding was determined to be more than minor because the finding was similar to IMC 0612, Appendix E, Example 4.b, and resulted in operator intervention to change reactor power to maintain reactor power at a stable value. Therefore, the performance deficiency impacted the Initiating Events cornerstone objective to limit the likelihood of those events that upset plant stability. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the Initiating Events cornerstone. While the finding increased the likelihood of a reactor trip, it did not increase the likelihood that mitigation equipment would not be available, and therefore, the inspectors determined the finding to be of very low safety significance. The finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a), because the licensee did not appropriately plan the work activity consistent with nuclear safety, incorporating risk insights, job site condition, or the need for planned contingencies, compensatory actions and abort criteria. Specifically, licensee personnel did not adequately research the impact of a circuit card's removal and reinsertion into the control circuitry for APRM 'A,' on other related systems contributing directly to an unplanned power transient on the reactor. (Section 4OA3.4)

- Green. A finding of very low safety significance was self-revealed on June 21, 2009, for the failure to adequately implement the requirements of Nuclear Operating Procedure (NOP)-WM-4300, Order Execute Process. Specifically, a supervisor authorized work order steps to be performed out of sequence on level switches for the moisture separator reheaters (MSR). The failure to perform steps in order led to some steps being missed and ultimately to a main turbine trip and associated reactor scram. The licensee entered this item into their corrective action program as CR 09-60855. The licensee's immediate actions included response to the reactor scram and formation of a troubleshooting team to conduct a root cause investigation of the failure of the MSR level indicators.

The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of procedure quality and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability. Specifically, inadequate adjustment and calibration of the level switches following replacement resulted in a main turbine trip and reactor scram from full power. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the Initiating Events cornerstone. While the finding resulted in a reactor trip, it did not contribute to the likelihood that mitigation equipment would not be available, therefore, the inspectors determined the finding to be of very low safety significance. This finding has a cross-cutting aspect in the area human performance, resources per IMC 0305 H.2(c), because the licensee did not ensure that procedures were adequate to assure nuclear safety. Specifically, the generic instrumentation and control instruction and the work order for conducting maintenance on the moisture separator reheater level switches did not contain critical vendor information or guidance to reflect the significance of taking as-found data to support calibration of the replacement switches. (Section 4OA3.6)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance (Green) and associated non-cited violation of license condition 2.C.(6), Fire Protection, was self-revealed for the licensee's failure implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR). Specifically, the licensee failed to ensure that, "…the main floor [of the Diesel Generator Building] is protected by a total flooding carbon dioxide system for fire suppression." The licensee had installed a permanent modification to the carbon dioxide system for the diesel generator room, but had chosen not to conduct complete post modification testing. The failure to conduct a complete test resulted in a wiring error to go undetected. Testing after the system was placed in service identified that the system did not function as designed. Troubleshooting identified that Division 2 and 3 emergency diesel generators (EDG) pneumatic electric relays in the new control panel were cross-wired to Division 3 and 2 EDG fan relays, respectively, in the control box. As part of their corrective actions, the licensee re-labeled the wires correctly in the CO_2 panels and landed them on their appropriate terminals. The licensee entered this issue into their corrective action program as CR 09-60866. The licensee's immediate corrective actions included placing the system in lockout and notifying all fire team personnel of the manual actions required to initiate CO_2 flow into the emergency diesel generator rooms.

  The finding was determined to be more than minor because, if left uncorrected, the inability of the EDG automatic fire suppression system to perform its function would become a more significant safety concern. Specifically, a fire in the Division 2 EDG room would not have been protected by adequate automatic fire suppression and it would render the Division 3 EDG inoperable. Similarly, a fire in the Division 3 EDG room would not have been protected by adequate automatic fire suppression and it would render the Division 2 EDG inoperable. The inspectors concluded this finding was associated with the Mitigating Systems cornerstone. In accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings,” Table 3b, the inspectors determined that the finding degraded the fire protection defense-in-depth strategies. Therefore, screening under IMC 0609, Appendix F, “Fire Protection Significance Determination Process,” was required. Using Part 1 of the Fire Protection SDP Phase 1 Worksheet in Manual Chapter 0609, Significance Determination Process, the performance issue was determined to be in the fixed fire protection systems category based on the fixed fire suppression systems being degraded. This finding did not screen as very low safety significance (Green) in the Phase 1 analysis and a Phase 2 analysis under IMC 0609 Appendix F was required.

A regional senior reactor analyst evaluated this finding and assumed the fire frequency to be 3.0E-2 for the EDG rooms based on the licensee’s IPEEE (Individual Plant Examination – External Events). Considering the fire frequency and remaining mitigating capability in the event of a plant transient, the senior reactor analyst determined that the risk associated with this finding was less than 1.0E-06. Therefore, this finding was determined to be best characterized as very low safety significance (Green). This finding has a cross-cutting aspect in the area of human performance, decision making, per IMC 0609 H.1(b), because the licensee’s decisions did not demonstrate that nuclear safety was an overriding priority. Specifically, the licensee chose to minimize system unavailability time over performing a full and complete post-maintenance test on a newly installed EDG CO_2 control system, a test that would have identified the wiring error. (Section 1R18.2)

- Green. A finding of very low safety significance (Green) and associated non-cited violation of Technical Specification 5.4.1 was self-revealed when the licensee failed to follow Nuclear Operating Procedure (NOP)-WM-3001; Work Management PM Processes. Specifically, step 4.5.5 of NOP-WM-3001 states, “If a General Nuclear Preventative Maintenance (GNPM) Order cannot be completed as planned due to ... replacement of a failed or degraded component, then the MWC Supervisor shall take appropriate actions in accordance with the flow diagram in Attachment 6 ...” During the performance of work order 200297036, for safety related 480-V breaker EF1A03, the supervisor directed a 4-point switch be replaced as part of the work order; however, no evaluation of the change in scope was completed and a CR was not written as required by Attachment 6 of NOP-WM-3001. The failure to evaluate the replacement lead to the loss of power to a number of safety related components. The licensee entered this item into their corrective action program as CR 09-63681. The licensee’s immediate action included entry into the appropriate technical specifications, restoration of the lost electrical power bus and restoration of emergency core cooling systems which were made inoperable as a result of the power loss.

  The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of human performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the inadequate work planning caused a loss of electrical power to bus EF-1-A, the safety-related 480 V power supply to Division 1 components placing the plant in an orange probabilistic safety analysis risk condition. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings," Table 2, the inspectors determined that core decay heat removal was degraded. Using Table 4a, "Characterization Worksheet for IE, MS, and BI Cornerstones," the inspectors assessed the finding as having very low safety significance (Green) because no loss of safety system function occurred and no loss of function of a single train occurred for greater than its TS-allowed outage time. This finding has a cross-cutting aspect in the area of human performance, work control per IMC 0609 H.3(b) because the licensee did not plan and coordinate work activities consistent with nuclear safety. Specifically, licensee personnel failed to plan and coordinate the replacement of an auxiliary switch in breaker EF1A03 thereby not incorporating the impact of the changes to the work scope or activity on the plant and human performance. (Section 4OA3.3)

B. Licensee-Identified Violations

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

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period at 100 percent power. On July 23, 2009, the operators reduced power to 37 percent power and returned to full power on July 25, 2009, and remained at full power for the remainder of the inspection period. The down power was driven by Technical Specification (TS) requirements due to a hydraulic fluid leak on the turbine bypass valve control system. The unit also performed a rod sequence exchange while down in power.

1. REACTOR SAFETY

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

1R01 Adverse Weather (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 Final Safety Analysis Report (UFSAR) 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, was verified to be in operation where applicable. The inspectors specifically reviewed the preparations for operation of the auxiliary boiler systems and building heating systems and the preparations to provide a temporary heating system for the enclosure around the cooling tower level control valve enclosure. The inspectors also reviewed corrective action program (CAP) items to verify that the licensee was identifying adverse weather issues at an appropriate threshold and entering them into their CAP in accordance with station corrective action procedures. Specific documents reviewed during this inspection are listed in the Attachment.

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

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Division 1 Emergency Diesel Generator (EDG) starting air and jacket cooling; and
- Standby liquid control.

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, UFSAR, TS requirements, outstanding work orders (WOs), 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.

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

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On September 25, 28, and 29, 2009, the inspectors performed a complete system alignment inspection of the high pressure core spray (HPCS) system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee’s probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment line ups, electrical power availability, system pressure and temperature indications, as appropriate, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment.

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

b. Findings

No findings of significance 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 Zone OIB-1; Intermediate Building 574' Elevation and Pipe Chase 585' Elevation;
- Fire Zone OIB-3; Intermediate Building 620'-6" Elevation;
- Fire Zone OIB-4; Intermediate Building 654'-6" Elevation;
- Fire Zone OIB-5; Intermediate Building 682' Elevation;
- Fire Zone OCC-1; Control Complex 574' Elevation; and
- Fire Zone OCC-2, Control Complex 599' Elevation.

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

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

b. Findings

No findings of significance were identified.

1R06 Flooding Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed and observed inspection of selected electrical safety vaults that were subject to flooding and contained cables whose failure could disable risk-significant equipment. The inspectors reviewed the WOs and inspection schedule detailing inspection periodicity for all underground vaults. The inspectors reviewed drawings and site plans to identify locations of underground vaults containing risk-significant cables. The inspectors also reviewed the licensee's corrective action documents with respect to discrepancies identified during the inspections to verify the adequacy of the corrective actions. The specific documents reviewed are listed in the Attachment to this report.

The inspectors performed a direct observation of open vaults during the licensee's inspection. The inspectors verified that the cables are not submerged in water; verified the integrity of the cables and support structures; verified that deficiencies were entered into the CAP; and verified that provisions for dewatering are sufficient to ensure that cables will not become submerged.

This inspection constituted one sample of cables located in underground vaults as defined in IP 71111.06-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

On July 27, 2009, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

- licensed operator performance;
- crew's clarity and formality of communications;
- ability to take timely actions in the conservative direction;
- prioritization, interpretation, and verification of annunciator alarms;
- correct use and implementation of abnormal and emergency procedures;
- control board manipulations;
- oversight and direction from supervisors; and
- 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 sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues involving the following systems:

- EDG CO_2 system during the month of July 2009 (see Section 1R18 for additional information on the installation of a permanent modification to the EDG CO_2 systems);
- service and instrument air during the week of September 14, 2009;
- residual heat removal (RHR) during the week of August 24, 2009; and
- main feed water system during the week of September 14, 2009.

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

b. Findings

No findings of significance 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:

- EDG Division 1 lube oil system check valve 1R47F0512A replacement on July 7, 2009; and
- HPCS logic system functional test performed on line on July 20, 2009.

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

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

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- control of leakage from valve EP-101 for rod 26-47 during the week of August 3, 2009; and
- Operational Decision Making Instruction for reactor feed booster pump 'A' during the week of September 21, 2009.

The inspectors selected these potential operability issues based on the risk-significance of the associated components and systems. The inspectors evaluated the technical adequacy of the evaluations to ensure that TS operability was properly justified and the subject component or system remained available such that no unrecognized increase in risk occurred. The inspectors compared the operability and design criteria in the appropriate sections of the TS and Updated Safety Analysis Report (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 also reviewed a sampling of corrective action documents to verify that the licensee was identifying and correcting any deficiencies associated with operability evaluations. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Plant Modifications

a. Inspection Scope

The inspectors reviewed the temporary modification for the drywell close-out inspection issues associated with floor drain covers and temporary lighting strings.

The inspectors compared the temporary configuration changes and associated 10 CFR 50.59 screening and evaluation information, against the design basis, the UFSAR, and the TS, as applicable, to verify that the modification did not affect the operability or availability of the affected system(s). The inspectors also compared the licensee's information to operating experience information to ensure that lessons learned from other utilities had been incorporated into the licensee's decision to implement the temporary modification. The inspectors, as applicable, performed field verifications to ensure that the modifications did not impact the operability of any interfacing systems. Lastly, the inspectors discussed the temporary modification with operations, engineering, and training personnel to ensure that the individuals were aware of how extended operation with the temporary modification in place could impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this document.

This inspection constituted one temporary modification sample as defined in IP 71111.18-05.

b. Findings

No findings of significance were identified.

.2 Permanent Plant Modifications

a. Inspection Scope

The engineering design package for the CO_2 panel modification was reviewed and selected aspects were discussed with engineering personnel. This document and related documentation were reviewed for adequacy of the associated 10 CFR 50.59 safety evaluation screening, consideration of design parameters, implementation of the modification, post-modification testing, and relevant procedures, design, and licensing documents were properly updated. The inspectors observed ongoing and completed work activities to verify that installation was consistent with the design control documents. The licensee upgraded the Unit 1 CO_2 fire suppression system control panels for all three diesel generator rooms. As part of the upgrade, the licensee replaced circuit boards in the fire suppression control panels with redundant panel control units, each capable of independently controlling each of the three systems separately. This would allow control of each system, without disabling all three systems if a problem develops in a control unit. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

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

b. Findings

Introduction: A finding of very low safety significance (Green) and associated non-cited violation of license condition 2.C.(6), Fire Protection, was self-revealed for the licensee's failure to implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR). Specifically, the licensee failed to ensure that "…the main floor [of the Diesel Generator Building] is protected by a total flooding carbon dioxide system for fire suppression."

Description: The licensee replaced all of the station's EDG CO_2 control panels in December 2008. Division 1 and 2 EDGs are the onsite emergency AC power sources and are credited for post-fire safe shutdown. The Division 3 EDG is dedicated to the HPCS pump but is not credited for post-fire safe shutdown.

On January 15, 2009, the licensee completed a post-installation test for all three EDG CO_2 panels in accordance with the approved WO (Order 200194345). The licensee chose not to conduct a complete CO_2 system test to limit the diesel generators' total unavailability time. The decision was based on the belief that portions of the system that were not affected by the system modification did not need to be tested, specifically the EDG HVAC system. To complete the as-described post-installation test, the licensee lifted the wires that connect the CO_2 panel to the associated HVAC systems to minimize EDG unavailability time. The test identified in the WO passed and the panels were placed into service on January 25, 2009. The modified CO_2 system panel's performance was verified to be correct by ensuring that appropriate contacts operated correctly when a CO_2 initiation signal was placed on the system.

On February 4, 2009, the licensee conducted a routine performance test of Division 3 EDG, which produced unsatisfactory results. During the performance test, while simulating a fire in the Division 3 EDG room, the associated HVAC system did not trip. It was unknown at that time what caused the HVAC system to not shutdown. As a precautionary measure, all three EDG CO_2 panels were placed in 'Lockout' (this function only allows manual initiation of the CO_2 system) until troubleshooting could be performed. The operators of the system did not know at the time that a software problem existed which prevented proper operation of the system in the 'Lockout' mode and a manual initiation from the CO_2 suppression system control panels would not have functioned correctly for any of the EDG rooms as the system was then configured.

Troubleshooting found that Division 2 and 3 EDG pneumatic electric relays in the control panel were cross-wired to Division 3 and 2 EDG fan relays, respectively, in the control box. It was noted at that time that the wires in the EDG CO_2 panels were labeled incorrectly, causing the wires to be swapped between the two divisions. Perry's Division 1 EDG was not affected by this wiring error which the licensee's root cause team determined was likely made during the post-installation test. In this condition, an automatic CO_2 system initiation would not have allowed the CO_2 system to function as designed in the associated EDG room and would have caused the other EDG to become inoperable. Specifically, in this wiring configuration for Division 2 and 3 EDGs, a fire in the Division 2 EDG would have tripped the fans and shut the dampers in the Division 3 EDG room and vice versa.

Further system testing revealed that with the panels in 'Lockout,' a manual initiation of the CO_2 system was not possible due to a software error in the new system. While in this condition, the licensee was unaware of its inability to manually use the CO_2 suppression system control panel to initiate fire suppression in any of the EDG rooms.

Perry Operations Manual, FTI-F0036, Post-Maintenance Testing Manual, step 5.1.1.1 states, "The test shall verify the ability of the affected component, system, or structure to perform its intended function." And step 5.1.1.3 states, "The test shall verify no new or related deficiencies have been created as a result of the maintenance activity." Contrary to the above, the licensee did not verify the ability of the CO_2 system to perform its intended function and did not verify there were no new deficiencies in the system.

As part of their corrective actions, the licensee re-labeled the wires correctly in the CO_2 panels and landed them on their appropriate terminals. The licensee also contacted the vendor of the CO_2 panels to assist in troubleshooting the software error in the system. The licensee informed operations and fire brigade personnel of the inability to initiate the CO_2 system into any of the EDG rooms through the new control panel. A back-up method was always available to the licensee and had in fact been the normal configuration prior to upgrading the system. The licensee will perform full functional tests of each EDG CO_2 system prior to placing the new system in service.

<u>Analysis</u>: The inspectors determined that the failure to perform a post-maintenance test that thoroughly verified the ability of the CO_2 system to perform its intended function was contrary to FTI-F0036, Post-Maintenance Test Manual, Revision 7, and was a performance deficiency.

The finding was determined to be more than minor because, if left uncorrected, the inability of the EDG automatic fire suppression system to perform its function would become a more significant safety concern. Specifically, a fire in the Division 2 EDG room would not be protected by adequate automatic fire suppression and it would render the Division 3 EDG inoperable. Similarly, a fire in the Division 3 EDG room would not be protected by adequate automatic fire suppression and it would render the Division 2 EDG inoperable. The inspectors concluded this finding was associated with the Mitigating Systems cornerstone.

In accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 3b, the inspectors determined that the finding degraded the fire protection defense-in-depth strategies. Therefore, screening under IMC 0609, Appendix F, "Fire Protection Significance Determination Process," was required. Using Part 1 of the Fire Protection SDP Phase 1 Worksheet in Manual Chapter 0609, Significance Determination Process [SDP], the performance issue was determined to be in the fixed fire protection systems category based on the fixed fire suppression systems being degraded. Specifically, a fire protection CO_2 actuation signal for Division 2 and 3 EDG rooms would not result in a trip of the room ventilation fans and isolation of the room dampers. Thus the CO_2 system for the EDG rooms would not have functioned as designed.

The degradation rating for this finding was high based on Division 2 and 3 EDGs being degraded such that no fire protection benefit existed given that the respective ventilation systems for the EDGs would not have functioned as designed and manual initiation of the fire suppression systems would not have functioned. This finding did not screen as very low safety significance (Green) in the Phase 1 analysis and a Phase 2 analysis under IMC 0609 Appendix F was required.

A regional senior reactor analyst evaluated this finding and assumed the fire frequency to be 3.0E-2 for the EDG rooms based on the licensee's IPEEE (Individual Plant Examination – External Events). The Transients worksheet from the Risk-Informed Notebook for Perry was assumed to correspond to the initiating event whose characteristics most closely resembled the impact of an assumed fire in the diesel generator rooms. High pressure core spray was not credited because its safety function was assumed to be unavailable. Considering the fire frequency and remaining mitigating capability in the event of a plant transient, the senior reactor analyst determined that the risk associated with this finding was less than 1.0E-06. Therefore, this finding was determined to be best characterized as very low safety significance (Green).

This finding has a cross-cutting aspect in the area of human performance, decision making, per IMC 0609 H.1(b), because the licensee's decisions did not demonstrate that nuclear safety was an overriding priority. Specifically, the licensee chose to minimize system unavailability time and as a result did not perform a full and complete post-maintenance test on a newly installed EDG CO_2 control system which would verify complete system functionality.

Enforcement: License condition 2.C.(6) required the licensee to implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (FSAR) and as approved through Safety Evaluation Reports dated May 1982. Updated Safety Analysis Report, section 9A.4.5, Diesel Generator Building, states that, "…the main floor is protected by a total flooding carbon dioxide system for fire suppression."

Contrary to the above, from January 25, 2009, to July 15, 2009, the licensee failed to maintain in effect the provisions in the fire protection program which require that the EDG rooms are protected with a total flooding CO_2 system for fire suppression. Specifically, the ventilation system would not have aligned properly to ensure an adequate concentration of CO_2 for fire suppression in Division 2 and 3 diesel rooms. Also, the licensee did not have the capability to manually initiate the fire suppression system for any of the diesel rooms. The licensee entered this issue into their corrective action program as CR 09-60866. The licensee's immediate corrective actions included placing the system in lockout and notifying all fire team personnel that manual actions would be required to initiate CO_2 flow into the emergency diesel generator rooms. Because this violation was of very low safety significance and was entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000440/2009004-01).

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:

- annulus exhaust gas treatment system, Train 'A' following charcoal sample and replacement; and
- Division 1 EDG following lube oil check valve replacement.

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

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

b. Findings

No findings of significance 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:

- standby liquid control Train 'A', pump and valve operability in-service testing on July 7, 2009;
- HPCS logic system functional test on July 20, 2009 (routine);
- reactor core isolation cooling (RCIC) pump and valve operability in-service testing on August 6, 2009;
- diesel generator DG start and load Division 3, routine testing on September 15, 2009; and
- lower airlock door seal routine testing during the week of September 21, 2009.

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

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

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three routine surveillance testing samples and two in-service testing samples as defined in IP 71111.22, Sections -02 and -05.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on July 17, 2009, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the control room (simulator) and the technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. As part of the inspection, the inspectors reviewed the drill package and other documents listed in the attachment.

These activities constituted one sample of an emergency drill observation as defined in IP 71114.06-05.

b. Findings

No findings of significance were identified.

.2 Training Observation

a. Inspection Scope

The inspector observed the simulator control room portion of a routine licensee emergency drill on August 18, 2009, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator (PI) data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors reviewed the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the CAP. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the Attachment to this report.

This inspection constituted one sample of a training observation as defined in IP 71114.06-05.

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically Significant Areas (71121.01)

.1 Plant Walkdowns

a. Inspection Scope

The inspectors reviewed the licensee's physical and programmatic controls for highly activated and/or contaminated materials (non-fuel) stored within the spent fuel pool or other storage pools.

This inspection constituted one sample as defined in IP 71121.01-5.

b. Findings

No findings of significance were identified.

.2 Radiation Work Permit and Document Reviews

a. Inspection Scope

The inspectors reviewed the radiation work permits (RWPs) and work packages used to access these areas and other high radiation work areas. The inspectors assessed the work control instructions and control barriers specified by the licensee.

This inspection supplemented samples previously documented in Inspection Report 05000440/2009002.

b. Findings

No findings of significance were identified; however, two violations of minor safety significance were identified by the inspectors.

1. On April 20, 2009, work was ongoing on valve 1E12F0010, which is the suction header valve on the RHR system. The area was appropriately posted and controlled as a locked high radiation area (LHRA). Workers on this job were issued electronic dosimeters (EDs) with inappropriate alarm set points. Upon initial entry into the work zone, the workers received dose alarms on their EDs. At this point, the workers followed plant procedures by stopping work and immediately exiting the area, and a radiation protection (RP) technician escorted the radiation workers to the "low dose area" to check their ED alarms. The RP technician notified his immediate supervisor, who in turn notified the RP superintendent and the station radiation protection manager (RPM). The RP superintendent and RPM assessed the cause and consequences of the ED alarms, assessed the radiological work environment, and assessed the radiological controls that were in place for compliance to plant TS for high radiation area controls. Once

compliance to the plant TS was verified and with all pertinent factors considered, the stop-work restriction was lifted and work resumed with the EDs in a dose alarm status, and with compensatory measures in place (i.e., time-keeping and direct RP technician coverage).

Allowing work to continue with the EDs in dose alarm mode effectively removed the alarm functions and caused the EDs to become the functional equivalent of a self-reading, non-alarm dosimeter, which was not recognized by the licensee. Perry Nuclear Power Plant NOP-OP-4101 Section 4.6, states, in part, that alarming direct reading dosimeter (EDs) with appropriate alarming augmentation are required for work in LHRAs. The failure to meet this requirement constitutes a violation of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. The event has been addressed by the licensee and is documented in the licensee's CAP in CR 09-57546.

2. Nuclear Operating Procedure, NOP-OP-4107, Section 4.1.4, required that the radiation worker is responsible for reading, understanding and adhering to RWP requirements and the applicable as-low-as-reasonably-achievable (ALARA) Plan. Procedure NOP-OP-4107, Section 4.3.2.4 stated that ALARA Plans will be considered part of the RWP for review, understanding, and reference prior to commencing work. On April 20, 2009, a job briefing was conducted regarding anticipated dose gradients within the body of Valve 1E12F0010, which is the suction header valve on the RHR system. This briefing led to a decision to use multiple EDs on the workers to more accurately monitor and record their dose. The ALARA Plan for RWP 096056 stated that the valve work was to be stopped and the ALARA Plan and RWP revised if dosimetry relocation, multi-badging or extremity monitoring were required. The work crew was briefed on the radiological hazards prior to beginning in field work activities. However, the failure of the licensee to revise the ALARA Plan and RWP 096056 prior to continuing work on Valve 1E12F0010 using multiple EDs is a violation of minor significance and is not subject to enforcement action in accordance with the NRC's Enforcement Policy. The event has been addressed by the licensee and is documented in the licensee's CAP in CR 09-57549.

2OS2 <u>As-Low-As-Reasonably-Achievable Planning and Controls</u> (71121.02)

.1 <u>Radiological Work Planning</u>

a. <u>Inspection Scope</u>

The inspectors compared the results achieved (including dose rate reductions and person-rem used) with the intended dose established in the licensee's ALARA planning for selected on-line work activities from 2008 and 2009. Reasons for inconsistencies between intended and actual work activity doses were reviewed.

This inspection supplements the activities reported in IR 05000440/2009-003 and completes one required sample as defined in IP 71121.02-5.

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

This inspection constituted one optional sample as defined in IP 71121.02-5.

The licensee's post job (work activity) reviews were evaluated to verify that identified problems were entered into the licensee's CAP.

This inspection constituted one optional sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

.2 Verification of Dose Estimates and Exposure Tracking Systems

a. Inspection Scope

The licensee's process for adjusting exposure estimates or re-planning work (when unexpected changes in scope, emergent work, or higher than anticipated radiation levels were encountered) was evaluated. This included determining whether adjustments to estimated exposure (intended dose) were based on sound RP and ALARA principles or whether they resulted from failures to adequately plan or to control the work. The frequency of these adjustments was reviewed to evaluate the adequacy of the original ALARA planning process.

This inspection constituted one required sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

.3 Declared Pregnant Workers.

a. Inspection Scope

The inspectors reviewed dose records of declared pregnant workers for the current assessment period to verify that the exposure results and monitoring controls employed by the licensee complied with the requirements of 10 CFR Part 20.

This inspection constituted one required sample as defined in IP 71121.02-5.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Occupational Exposure Control Effectiveness

a. Inspection Scope

The inspectors sampled licensee submittals for the Occupational Radiological Occurrences PI for the period from the third quarter 2008 through the second quarter 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, 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 its data review, and the results of those reviews. The inspectors independently reviewed electronic dosimetry dose rate and accumulated dose alarm 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 selected LHRA 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 radiological occurrences sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures PI for the period from the third quarter 2008 through the second quarter 2009, to determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, and NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73" definitions and guidance, were used. The inspectors reviewed the licensee's operator narrative logs, operability assessments, maintenance rule records, maintenance WOs, issue reports, event reports and NRC Integrated Inspection Reports for the period of July 1, 2008, to June 30, 2009, 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 safety system functional failures sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.3 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Transients per 7000 Critical Hours PI for the period from the third quarter 2008 through the second quarter 2009. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, was used. The inspectors reviewed the licensee's operator narrative logs, issue reports, maintenance rule records, event reports and NRC Integrated Inspection Reports for the period of July 1, 2008, to June 30, 2009, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned transients per 7000 critical hours sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.4 Reactor Coolant System (RCS) Leakage

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Leakage PI for the period from the fourth quarter 2008 through the second quarter of 2009 to determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 5, were used. The inspectors reviewed the licensee's operator logs, RCS leakage tracking data, issue reports, event reports and NRC Integrated Inspection Reports for the period of October 2008 through June 2009 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.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Items Entered Into the CAP

a. Inspection Scope

As part of the various baseline IPs discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's CAP at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Attributes reviewed included: the complete and accurate identification of the problem; that timeliness was commensurate with the safety significance; that evaluation and disposition of performance issues, generic implications, common causes, contributing factors, root causes, extent-of-condition reviews, and previous occurrence 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 attached List of Documents Reviewed.

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 of significance 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 of significance were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of January 1, 2009, through June 30, 2009, although some examples expanded beyond those dates where the scope of the trend warranted.

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

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

b. Findings

No findings of significance were identified.

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

.1 Down Power to 37 Percent Due to Turbine Bypass Valve Control System Electro-Hydraulic Control System (EHC) Fluid Leak

a. Inspection Scope

The inspectors reviewed the licensee's response to an EHC fluid leak on the turbine bypass valve control system on July 23, 2009, and July 24, 2009. The inspectors evaluated the initiating cause of the down power and the personnel response requiring more than routine operator actions. The inspectors determine that the response was appropriate and in accordance with TS, procedures, and training. The inspectors reviewed operator logs, plant computer data, strip charts, and other data after stable plant conditions were achieved. The inspectors also monitored subsequent power ascension activities. Documents reviewed are listed in the Attachment.

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

b. Findings

No findings of significance were identified.

.2 Valve Leak on Rod 26-47

a. Inspection Scope

The inspectors reviewed the licensee's response to a body-to-bonnet leak on valve EP-101. The leaking valve is in the control rod drive cooling portion of piping which supports control rod 26-47. The leak was discovered visually and did not challenge operability of the rod. The main concern was with spray effects to other instruments in the vicinity of the leak. The licensee controlled the spray and made initial efforts to reduce the leak rate by torquing the body-to-bonnet bolts to maximum torque. The efforts to reduce the leak were unsuccessful. Repair work on the leak was scheduled for October 2009. Documents reviewed are listed in the Attachment.

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

b. Findings

No findings of significance were identified.

.3 Maintenance Errors Cause Loss of Div 1 Emergency Core Cooling System (ECCS) Electric Power

a. Inspection Scope

The inspectors reviewed the licensee's response to a loss of the Division 1 safety buses that occurred on August 25, 2009, when operators were restoring from preventative maintenance on two safety-related breakers. A tie-breaker tripped when fuses were installed which caused a loss of the Division 1 480-volt safety-related bus. The underlying cause of this breaker trip was a mis-positioned auxiliary switch during the preventative maintenance activity

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

b. Findings

Introduction: A finding of very low safety significance (Green) and associated non-cited violation of TS 5.4.1 was self-revealed for the licensee failing to follow step 4.5.5 of NOP-WM-3001; Work Management PM Processes.

Description: On August 25, 2009, at 2350, during restoration from preventative maintenance on two safety-related breakers (one 4160 V and one 480 V), through a sequence of events described below, when auxiliary operators installed fuses into the Division 1 tie-breaker EF1B13, safety-related bus EF1A, which powers emergency service water (ESW) 'A' and emergency core cooling (ECC) 'A,' tripped. Several additional Division 1 safety-related components were also rendered inoperable, including RHR 'A' and its associated low pressure core injection (LPCI) function; low pressure core spray (LPCS); RCIC; and Division 1 EDG. The loss of all Division 1 ECC systems caused the plant to enter a significant (orange) plant risk condition.

A review by the licensee of personnel actions leading up to the event identified that on August 24, 2009, during performance of WO 200297036 on 480 V breaker EF1A03, maintenance electricians performing the task noted discoloration on the contacts of an auxiliary 4-point switch. After consulting with their supervisor, a decision was made to replace the auxiliary switch. The supervisor directed the scoping of the replacement into the WO, but no further evaluation of the change in scope was completed and no procedure for replacing the switch was used. During reassembly of the breaker, following replacement of the 4-point auxiliary switch, the electricians unknowingly mis-positioned an 8-point switch that had been disassembled during the work activity.

Step 4.5.5 of NOP-WM-3001; Work Management PM Processes, states, “If a General Nuclear Preventative Maintenance (GNPM) Order cannot be completed as planned due to a needed corrective repair, rework or replacement of a failed or degraded component, then the MWC Supervisor shall take appropriate actions in accordance with the flow diagram in Attachment 6 for processing of the GNPM Order.” Attachment 6 of NOP-WM-3001 directs a CR to be written when critical components are required to be replaced. Evaluation of the CR will advise a resolution to the WO. Contrary to the above, the supervisor did not write a CR for resolution of the WO which required replacement of a critical component (the 4-point auxiliary switch).

Electrical maintenance personnel completed their portion of the WO and turned breaker EF1A03 over to operations for post-maintenance testing. Operations used procedure SOI-R23, 480 Volt Load Centers, as part of the post-maintenance testing. Procedure SOI-R23 step 7.15.4 states, “If control power fuses are removed, then perform the following: install control power fuses.” The control power fuses for breaker EF1A03 had not been removed and were, in fact installed. However, the operators noted that the fuses for tie-breaker EF1B13 were removed and, without checking with their supervision or questioning their actions, mistakenly installed the fuses in EF1B13 (the fuses for EF1B13 had been removed by another procedure).

Operators installing fuses into the incorrect breaker was contrary to procedure SOI-R23 and was the second performance deficiency which led to the event. Breaker EF1B13 tripped open upon the installation of the fuses because of the improper configuration of the 8-point auxiliary switch in breaker EF1A03. The mis-positioned 8-point auxiliary switch set up the automatic trip logic for EF1B13 that was satisfied when the operators installed the fuses into breaker EF1B13, causing it to open and resulting in a loss of all power to bus EF-1-A.

Power to the 480-V safety-related bus EF-1-A was restored on August 26, 2009, at 0449; RCIC was declared operable at 1141 on August 26, 2009; LPCS was declared operable on August 26, 2009, at 1504; and RHR 'A' was declared operable for LPCI on August 28, 2009, at 1209.

Analysis: The inspectors determined that replacement of an auxiliary switch in breaker EF1A03 during a preventative maintenance task without following the steps in NOP-WM-3001; Work Management Processes, to generate a proper work order and associated condition report was the significant performance deficiency which caused this event. An additional performance deficiency involving personnel errors while installing fuses contributed to the loss of power to safety-related Division 1 electrical bus EF-1-A but would not have caused the event by itself.

The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of human performance and affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, an 8-point auxiliary switch in breaker EF1A03 was incorrectly configured and caused a loss of electrical power to bus EF-1-A, the safety-related 480 volt power supply to Division 1 components, including ESW 'A' and ECC 'A', rendering them inoperable and placing the plant in an orange probabilistic safety analysis risk condition.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 2, the inspectors determined that core decay heat removal was degraded. Using Table 4a, "Characterization Worksheet for IE, MS, and BI Cornerstones," the inspectors assessed the finding as having very low safety significance (Green) since no loss of safety system function occurred and no loss of function of a single train occurred for greater than its TS-allowed outage time.

This finding has a cross-cutting aspect in the area of human performance, work control per IMC 0609 H.3(b), because the licensee did not plan and coordinate work activities consistent with nuclear safety. Specifically, replacing an auxiliary switch in breaker EF1A03 during a preventative maintenance task without an approved WO did not incorporate the impact of the changes to the work scope or activity on the plant and human performance.

Enforcement: Section 5.4.1 of the TS states, in part, that "Written procedures shall be established, implemented, and maintained covering the following activities: The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978." Paragraph 9.a. of this Regulatory Guide states, in part, that procedures for performing maintenance that can affect the performance of safety-related equipment shall be prepared and activities shall be performed in accordance with these procedures. The licensee established NOP-WM-3001, Work Management PM Process as the implementing procedure for execution of preventative maintenance orders. Attachment 6 of NOP-WM-3001 directs a CR to be written when critical components are required to be replaced. Evaluation of the CR will advise a resolution to the WO.

Contrary to the above, on August 25, 2009, the licensee failed to follow NOP-WM-3001 by not writing a CR when a critical component was required to be replaced. Specifically, the licensee failed to generate a CR when a 4-point switch, a critical component, for safety related 480 V breaker EF1A03 needed to be replaced. The licensee entered this item into their corrective action program as CR 09-63681. The licensee's immediate action included entry into the appropriate technical specifications, restoration of the lost electrical power bus and restoration of emergency core cooling systems which were made inoperable as a result of the power loss. Because this violation was of very low safety significance and was entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy (NCV 05000440/2009004-02).

.4 Unexpected Half Scram Due to Faulty Troubleshooting Plan

a. Inspection Scope

The inspectors reviewed the licensee's response to an unexpected half scram signal which was generated during troubleshooting of a failure of the 'A' APRM instrument.

Technicians troubleshooting the failed instrument caused a power spike which also tripped 'E' APRM causing the half scram signal and requiring operator action to control reactor power. Documents reviewed are listed in the Attachment.

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

b. Findings

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 the licensee's failure to provide an appropriate troubleshooting plan for APRM 'A'.

Description: On September 2, 2009, while troubleshooting inoperable APRM 'A,' the 15-Vdc (volt direct current) voltage regulator card in the control circuitry for the 'A' APRM was removed and reinstalled. Unknown to the technicians and the supervisors who developed the troubleshooting plan, removal and reinstallation of the 15-Vdc regulator card into the circuitry should be done while depressing the power supply (PS) 23 reset button. Power supply 23 provided power for the 15-Vdc voltage regulators for APRM 'A' and 'E'.

When the 15-Vdc card was reinstalled, a voltage spike caused PS 23 to trip resulting in a loss of power to APRMs 'A' and 'E' voltage regulator circuits. A reactor half scram was generated when APRMs 'A' and 'E' tripped. Operators reduced reactor power to allow the reactor recirculation water flow control valves to be locked to prevent an unnecessarily automatically run back through a flow control circuit had the APRM outputs drifted higher during the APRM outage. As a result of the power change, the operators entered Off-Normal Instruction (ONI)-C51, Unplanned Change in Reactor Power or Reactivity.

The 15-Vdc regulator card was suspected to be the component which caused the original failure of the APRM. When the new card arrived, the maintenance repair team, conducting the procedure to calibrate the replacement card, identified a note in the procedure that warned of the necessity to push the reset button while inserting or removing the card. This was the point at which the licensee recognized the fault which caused the half scram on September 2, 2009. Corrective actions planned include installation of a placard on the electronics drawers of the APRM units to ensure that the knowledge contained in the card calibration procedure is available to all maintenance personnel upon entry into the APRM control panels.

Analysis: The inspectors determined that the troubleshooting plan developed for inspection of the 15-Vdc regulating card did not adequately address the proper method to execute the task. The failure to use the proper method resulted in a half scram signal to the reactor protective system and required operator actions to control reactor power. The inappropriate troubleshooting plan was contrary to 10 CFR 50, Appendix B, Criterion V which requires in part that "activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances" and was a performance deficiency. The finding was determined to be more than minor because the finding was similar to IMC 0612, Appendix E, Example 4.b, and resulted in a reactor power transient requiring operator intervention to maintain the reactor power at a stable value. Therefore, this performance deficiency impacted the

Initiating Events cornerstone objective to limit the likelihood of those events that upset plant stability.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the initiating events cornerstone. Because the finding did not contribute to both the likelihood of a reactor trip and the likelihood that mitigation equipment would not be available, the inspectors determined the finding to be of very low safety significance (Green).

This finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a), because the licensee did not appropriately plan the work activity consistent with nuclear safety, incorporating risk insights, job site condition, or the need for planned contingencies, compensatory actions and abort criteria. Specifically, licensee personnel did not adequately research the impact of card removal and reinsertion into the control circuitry for APRM 'A' on other related systems contributing directly to an unplanned power transient on the reactor.

Enforcement: Criterion V, "Instructions, Procedures, and Drawings," of 10 CFR Part 50, Appendix B, requires, 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 September 2, 2009, the licensee's troubleshooting plan for "A" APRM, an activity affecting quality, was not appropriate to the circumstances. Specifically, the troubleshooting plan did not provide the proper steps for extracting and inserting the 15-Vdc voltage regulator card in the control circuitry for the 'A' APRM. The failure to properly remove and re-install the card resulted in a half scram with a subsequent requirement for operator action to maintain the plant's stability. The licensee entered the error into their corrective action program as CR 09-63991. The licensee's immediate actions included entry into an off-normal instruction to control reactor power while restoring the electric power supply to the effected APRMs and resetting the half scram signal. Because this violation was of very low safety significance and it was entered into the licensee's CAP via CR 09-63991, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2009004-03)

.5 Loss of Reactor Feed Booster Pump (RFBP) 'A'

a. Inspection Scope

The inspectors reviewed the licensee's response to a loss of RFBP 'A' which occurred on September 9, 2009, while the plant was operating at full power. The plant responded as designed when RFBP 'D' started upon the loss of the 'A' RFBP. The licensee developed guidelines for an Operational Decision Making Issue to instruct plant operators regarding the plant operating configuration without an available backup to the three running booster pumps and plant operations continued normally after the failure of RFBP "A".

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

b. Findings

No findings of significance were identified.

.6 Moisture Separator Reheater (MSR) Level Switch Maintenance Caused Unit Trip

a. Inspection Scope

During the second quarter inspectors conducted an initial review of the licensee's response to a main turbine trip and reactor scram which occurred on June 21, 2009, while the plant was operating at full power. The inspectors continued this review in the third quarter when the trip report and root cause were completed.

This event follow-up review does not constitute another sample as defined in IP 71153-05.

b. Findings

Introduction: A finding of very low safety significance (Green) was self-revealed on June 21, 2009, for the failure to adequately implement the requirements of NOP-WM-4300, Order Execute Process. Specifically, a supervisor authorized WO steps to be performed out-of-sequence on level switches for the moisture separator reheaters. This authorization directly contributed to a false high level signal that tripped the main turbine and caused a reactor scram on June 21, 2009.

Description: On June 21, 2009, the plant was operating in Mode 1 when an unplanned automatic reactor scram occurred due to turbine control valve fast closure activation of the reactor protection system. The cause of the main turbine trip and subsequent reactor scram was determined to be a high moisture separator reheater (MSR) water level signal.

The MSR level detectors were worked on during the spring outage of 2009. Supplemental work force conducted the outage maintenance on the level detectors that failed. Interviews of the site work force determined that a supervisor authorized performing steps in the WO out-of-sequence which directly resulted in the operators not completing a step to take "as-found data." By not completing the "as-found step," the calibrations of the level detectors were not completed correctly during the outage. Review of the WO, that controlled the maintenance, identified that the procedure did not include adequate directions to assure that "as-found data" would be taken by the workers to support post-installation calibration of the level detectors. After 1 month of operations at power, some of the improperly calibrated switches shifted to a false high level trip condition causing a turbine trip and associated reactor scram.

Proper calibration of the micro-switches would have prevented this event from occurring. The licensee entered this issue into the CAP (CR 09-60855) and conducted a full root cause evaluation. No corrective action reports were written by the work force during the outage to document the difficulties with the procedure to calibrate the level switches or the authorizations that were given by supervisors to complete the work. An industry operating experience report from Catawba was found by the licensee that directly reflected the necessity of taking as-found data to ensure proper post-corrective maintenance calibrations of the same type of micro-switches.

Analysis: The inspectors determined that maintenance performed on the MSR level micro-switches was contrary to the requirements of NOP-WM-4300, Order Execute Process, in that a supervisor authorized performing steps out-of-sequence which allowed some steps to not be completed at all and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of procedure quality and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability. Specifically, inadequate adjustment of the level switch following replacement resulted in a main turbine trip and reactor scram from full power.

The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 4a, for the initiating events cornerstone. While the finding resulted in a reactor trip, it did not contribute to the likelihood that mitigation equipment would not be available; therefore, the inspectors determined the finding to be of very low safety significance (Green).

This finding has a cross-cutting aspect in the human performance area, resources, per IMC 0305 H.2(c), because the licensee did not ensure that procedures are adequate to assure nuclear safety. Specifically, the generic instrumentation and control instruction and the WO for conducting maintenance on the MSR level switches did not contain critical vendor information or guidance to reflect the significance of taking as-found data to support calibration of the replacement switches.

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. The finding was of very low safety significance and was addressed in the CAP as CR 09-60855. The licensee's immediate actions included response to the reactor scram and formation of a troubleshooting team to conduct a root cause investigation of the failure of the MSR level indicators. This finding is identified as FIN 05000440/2009004-04.

.7 (Closed) Licensee Event Report (LER) 05000440/2009001-00, MSR High Level Signal Causes Turbine Trip and Reactor Protection System Actuation.

On June 21, 2009, the reactor protection system automatically actuated due to receipt of a turbine control valve fast closure signal. All control rods fully inserted and there were no complications during the shutdown. Reactor coolant pressure and level were maintained within expected parameters. The event was caused by an invalid high water level signal from MSR 1B instrumentation. The licensee determined that the direct cause for the invalid signal was incorrect adjustment and calibration of the switches during the previous refueling outage. The trip response was reviewed as an event follow-up. An associated self-revealed finding is documented in Section 4OA3.7. This LER is closed.

.8 (Closed) LER 05000440/2009002-00, Diesel Generator CO2 Fire Suppression Control Panel Mis-wiring Results in an Unanalyzed Condition

On June 22, 2009, the licensee discovered that a wiring error in a fire protection CO_2 panel resulted in a condition in which a fire protection CO_2 actuation signal for the Division 3 diesel generator room would cause the Division 2 diesel generator room ventilation supply fans to isolate and vice versa. The condition existed from

January 15, 2009, to June 23, 2009, when the wiring error was corrected. The root cause was determined to be an inadequate post-modification test which failed to identify the mis-wiring of two output wires from the diesel generator CO_2 fire suppression system control panel. An associated self-revealed NCV is documented in Section 1R18. This LER is closed.

4OA5 Other Activities

Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to the Site Vice President, Mr. Mark Bezilla, and other members of licensee management on October 7, 2009. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The preliminary results of the inspection of the licensee’s access control to radiologically significant areas, the ALARA planning and controls program, and the PI verification for occupational radiation safety, were discussed with FENOC Fleet Regulatory Affairs Director, Mr. G. Halnon and Perry Radiation Protection Manager Mr. P. McNulty, on July 10, 2009.
- Results of inspection activities associated with recent allegations regarding the licensee’s access control to radiologically significant areas with Mr. J. Pelsic and Mr. P. McNulty on August 11, 2009 via telephone.
- The preliminary results of the inspection of the licensee’s access control to radiologically significant areas were discussed with Perry Radiation Protection

Manager Mr. P. McNulty, on September 16, 2009, and Perry Regulatory Affairs Representative Mr. J. Pelcic on September 17, 2009, via telephone.

The inspectors confirmed that none of the potential report input discussed was considered proprietary.

4OA7 Licensee-Identified Violations

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

- Perry TS 5.7.1 states, in part, that each high radiation area shall be barricaded and conspicuously posted as a high radiation area. Contrary to the above, on March 11, 2009, the reactor water reject line to the main condenser was identified to have dose rates up to 235 mRem/hr at 30cm thereby creating a high radiation area in the main condenser hotwell. A violation of regulatory requirements occurred when the area was not effectively barricaded, controlled, and conspicuously posted as a high radiation area. This was identified in the licensee's CAP as CR 09-55159. Immediate corrective actions were to post and barricade the area in accordance with station procedures and applicable regulatory requirements. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised.

- Personnel entering high radiation areas must comply with Perry Plant TS 5.7.1, which states in part, that entry into such areas may be made after dose rate levels in the area have been established and personnel are aware of them. Contrary to the above, on March 15, 2009, a station employee entered an area of the Auxiliary Building elevation 599' that had actual radiation levels in excess of 100 mRem/hr without the requisite brief. The specific area was appropriately posted and controlled as a high radiation area. However, the individual was not made aware of the radiological conditions in the area prior to entry. No radiological exposures of significance occurred as a result of this event. This was identified by a RP technician and was documented in the licensee's CAP on CR 09-55466. Immediate corrective actions included the individual immediately exiting the area and the licensee assessing the radiological consequence of the entry. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
M. Bezilla, Vice President Nuclear
K. Krueger, Plant General Manager
A. Cayia, Director, Performance Improvement
M, Stevens, Director, Maintenance
D. Evans, Manager, Operations
J. Grabner, Director, Site Engineering
E. Gordon, Radiation Protection Superintendent
H. Hanson, Jr., Director, Work and Outage Management
P. McNulty, Radiation Protection Manager
P. New, Radiation Protection
J. Pelcic, Regulatory Affairs

NRC
D. Passehl, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Opened and Closed

05000440/2009004-01	NCV	Failure to Perform An Adequate Post-Maintenance Test Following Installation of New Emergency Diesel Generators Carbon Dioxide System Control Panels (Section 1R18.2)
05000440/2009004-02	NCV	Maintenance Errors Cause Loss of Division I ECCS Electrical Power (Section 4OA3.3)
05000440/2009004-03	NCV	Unexpected Half Scram Due to Faulty Troubleshooting Plan (Section 4OA3.4)
05000440/2009004-04	FIN	Moister Separator Reheater Level Switch Maintenance Caused Unit Trip (Section 4OA3.6)

Closed

05000440/2009001-00	LER	MSR High Level Signal Causes Turbine Trip and Reactor Protection System Actuation (Section 4OA3.8)
05000440/2009002-00	LER	Diesel Generator CO_2 Fire Suppression Control Panel Mis-Wiring Results in an Unanalyzed Condition (Section 4OA3.9)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather

Procedure PTI-Gen-P0026; Preparations for Winter Operation; Revision 5
Procedure PTI-Gen-P0027; Cold Weather Support System Startup; Revision 12
Procedure IOI-15; Seasonal Variations; Revision 17
Procedure NOP-WM-2001; Work Management Scheduling/Assessment/Seasonal Readiness Processes; Revision 8

1R04 Equipment Alignment

VLI-R46; Division 1 & 2 Diesel Generator Jacket Water Systems, Revision 4
VLI-R44; Division 1 & 2 Diesel Generator Starting Air, Revision 4
VLI-C41; Standby Liquid Control System; Revision 8
CR 09-63270; 1C41F0038A Found Unlocked; dated August 14, 2009

1R05 Fire Protection (Annual/Quarterly)

FPI-A-A02, "Periodic Fire Inspections," Revision 5
PAP-1910, "Fire Protection Program," Revision 19
PAP-0204, "Housekeeping/Cleanliness Control Program," Revision 22
FPI-OIB; Intermediate Building, Revision 5
FPI-OCC; Control Complex, Revision 7

1R06 Flooding Protection Measures

CR 04-04756; OE Re-evaluation – IEN 2002-12 Submerged Safety Related Electrical Cables; dated September 15, 2004
CR 02-00931; OE NRC IEN 2002-12 Submerged Safety Related Electrical Cables; dated March 28, 2002
CR 03-01436; Buried Cable Degradation Monitoring Program – NRC Resident Inquiry; dated March 21, 2003
CR 04-01042; Extended Delay in Inspection of Underground Cables; dated March 1, 2004
DWG 215-0711-00000; Manholes and Underground Duct Runs – Plans; Revision R
DWG D-216-016; Underground Duct Runs Div 1 to Emergency Service Water Pump House; Revision H
WO 20271754; Inspect, Dewater Safety Related Electrical and Communication Manholes; dated April 29, 2009
WO 200248121; Visual Inspection of Manholes; dated November 30, 2007
WO 200333849; Inspect and Dewater Manholes 3, 4 and 18; dated May 7, 2009
WO 200333795; Inspect, Dewater Safety Related Electrical and Communication Manholes; dated June 11, 2009
WO 200352613; Visual Inspection of Cables and Supports/ Record Sludge Levels of Safety Related Manholes 3 and 4 Division 1; dated July 10, 2009

NORM-ER-3112; Cable Monitoring; dated July 31, 2008
PY-CEI/NRR-3034L; Response to NRC Generic Letter 2007-01; dated May 8, 2007
SP-560-4549-00; Specification Class 1E Small Power and Control Cables; dated May 11, 1983

1R11 Licensed Operator Requalification Program

Simulator Discrepancies for Cycle 3; dated July 27, 2009
Simulator Exercise Guide OTLC-3058200903-PY-SGC2; Cycle 3 Evaluated Scenario C2; Revision 0
Simulator Examination Summary Sheet; dated July 27, 2009
Crew Competency Grading Worksheet; dated July 27, 2009
Overall Dynamic Simulator Individual Evaluation (SRO); dated July 27, 2009
Procedure PYBP-PTS-0005; Operator Continuing Training Program Administration; Revision 18

1R12 Maintenance Effectiveness

NOP-ER-3004; FENOC Maintenance Rule Program; Revision 1
PYBP-PES-0001; Perry Maintenance Rule Reference Guide; Revision 14
PAP-1125; Monitoring the Effectiveness of Maintenance Program Plan; Revision 8
Maintenance Rule Functions, Performance Criteria, and Classifications; Maintenance Rule Guidance document; dated January 11, 2008
Perry Residual Heat Removal System Health and Status Report 2009-1, January 1 through March 31, 2009
Perry Residual Heat Removal System Health and Status Report 2009-1, April 1 through June 30, 2009
List of CRs generated for the RHR System; January 2008 through August 2009
CR 09-56267; Attempt to Remove a RHR Motor with Coupling Bolts Still Attached; dated April 20, 2009
CR 09-58995; Procedure Change to Vent Piping between RHR to FW Return Isolation Valves; dated May 10, 2009
CR 09-56938; 1E12F0010 Improper Valve Indication; dated May 8, 2009
CR 08-48686; Performance Testing of RHR 'B' Loop; dated November 26, 2008
CR 09-57736; PY-PA-09-01: E12F0010 Hardware Disposition; dated April 30, 2009
CR 09-61658; RHR Pump A Discharge Pressure HI/LO, P601-20-G6 Alarm Received During LPCS Pump Operation; dated July 9, 2009
CR 09-58808; RHR 'A' High Pressure During ISI-B21-T1300-1; dated June 5, 2009
CR 09-57011; Shutdown Cooling Isolation Valve Has Abnormal Indication; dated April 13, 2009
CR 09-56953; QC ID: RHR 'A' Pump Motor Installed in Plant without Being Issued from Warehouse; dated April 9, 2009
CR 09-57964; After Pump Shutdown RHR B Discharge Pressure Low; dated April 25, 2009
CR 09-57118; RHR Pump 'A' Excessive Minimum Flow Run Time; dated April 11, 2009
CR 09-57354; Ran RHR Pumps B & C on Min Flow Greater Than One Hour During SVI-R43-T5367; dated April 16, 2009
WO 200337992; Evaluation of NRC Information Notice (IN) 2008-13
Perry Feedwater and Feedwater Leakage Control System Health Report 2009-2
CR 09-60869; Flow to RPV with MFP FCVS N27F110 and F010 Closed; dated June 21, 2009
CR 09-59655; HST Low Level Alarm Slow to Clear and Level Control Valve Slow to Respond, dated May 26, 2009
CR 08-43835; RFPT B Failed Lockout Suppressed Trip Test; dated July 27, 2008
CR 09-55206; Feedwater Check Valve 1N27F0559A Failed LLRT in 1R12; dated March 10, 2009

CR 08-44480; Water in Feed Pump Lube Oil System; dated August 7, 2008
CR 08-37457; Motor Feed Pump Oil Milky Appearance; dated March 29, 2008
CR 08-47924; Failed Goal within a Maintenance Rule (A)(1) Goal Monitoring Criteria; dated October 14, 2008

1R13 Maintenance Risk Assessments and Emergent Work Control

Division 1 Diesel Generator Outage (Yellow) Protected Equipment Posting Checklist; dated July 6, 2009
Tagout: PY-CYC-013; Clearance PY1-R43-0001; Div 1 Emergency Diesel Generator; dated July 7, 2009
WO 200373963; Replace Lube Oil Check Valve 1R470512A; dated July 6, 2009
WO 200261999; HPCS Logic System Functional Test; dated July 20, 2009
CR 09-60483; Excessive Fluid Coming From RB#5 During Div 1 DG Prestart Inspection; dated June 12, 2009
CR 09-690561; Div 1 DG Oil in Power Cylinder; dated June 15, 2009
Perry On-Line Probabilistic Risk Assessment; dated July 6, 2009, through July 12, 2009
Perry On-Line Probabilistic Risk Assessment; dated July 20, 2009, through July 26, 2009
Procedure PYBP-POS-5-11; Operations Work Control Unit (WCU) Guide; Attachment 5; Elevated Risk Work Authorization, Revision 9
Maintenance Excellence Pre-Job Briefing Checklist; dated July 20, 2009

1R15 Operability Evaluations

Operation with Reactor Feed Booster Pump (RFBP) 'A' Out of Service; 3 RFBP’s (B, C & D) Remain Available; Revision 0; dated 11 September 2009
Procedure NOP-OP-1010; Operational Decision Making; Revision 1

1R18 Permanent/Temporary Modifications

DWG 921-0601-0000; Reactor Building Floor and Equipment Drains West; Revision E
DWG 921-0001-0000; Typical Details or Accessories – Drainage; Revision E
Procedure PTI-G61-P0001; Drywell Floor Drain Sump Flow Capacity Test; Revision 2
CR 09-58994; Drywell Floor Drain Cover Has a Machined Rectangular Hole; dated May 9, 2009
CR 09-59279; Drywell Floor Drain Cover Has a Machined Rectangular Hole; dated May 9, 2009
CR 09-60923; Drywell Floor Drain Cover Non-Conforming Condition; dated June 23, 2009
WO 200369364; Replace Drywell Floor Drain Cover
CR 09-58976; Temporary Light String and Communications Cable Left Under Vessel After Demob; dated May 9, 2009
CR 09-59248; PY-PA-09-01 Process Compliance with Respect to CR 09-58976 Investigation; dated May 12, 2009
Procedure NOP-WM-4001; Foreign Material Exclusion; Revision 9
Procedure PAP-0202; Housekeeping/Cleanliness Control Program; Revision 22
Procedure NOP-CC-2003; Engineering Changes; Revision 14

1R19 Post-Maintenance Testing

WO 200317017; (18M) “A” Annulus Exhaust Gas Treatment Charcoal Adsorber Operability Test and Plenum Inspection
Procedure PY-SVI-M-15T3015; Annulus Exhaust Gas Treatment Charcoal Absorber Operability Test and Plenum Inspection; Revision 8
Carbon Sample Shipment Sample Number 670090

WO 200377145; Annulus Exhaust Gas Treatment System Train A Flow and Filter
Procedure PY-SVI-MIST1240A; Annulus Exhaust Gas Treatment System Train A Flow and Filter, Revision 6
Procedure GMI 0171; HECA Filter Bed Sampling (Canister Method); Revision 1
WO 200373963; Replace Lube Oil Check Valve 1R470512A; dated July 6, 2009
WO 200348988; (31D) Diesel Generator Start and Load Division 1; dated July 8, 2009
Procedure SI-R43-T1317; Diesel Generator Start and Load Division 1; Revision 14
CR 02-04254; Spare Check Valves Used on R47 System; dated November 11, 2002
CR 09-60692; Division 1 Diesel Engine Lube Oil Strainers are Mis-labeled; dated June 17, 2009
CR 09-60561; Div 1 DG Oil in Power Cylinder; dated June 15, 2009
CR 09-60483; Oil Coming From RB#5 During Div 1 DG Prestart Inspection; dated June 12, 2009
Cooper-Enterprises Service Information Memo; Lube Oil Check Valve Rework R&RV Engines; dated April 30, 2001
DWG 305-0353-00003; ISI Classification Boundary Piping Diagram – Standby Diesel Generator Lube Oil System; Revision 003
302-0353-00000; Standby Diesel Generator Lube Oil System P&ID; Revision S
302-0352-00000; Standby Diesel Generator Fuel Oil System P&ID; Revision GG
Vendor Manual Standby Diesel Generator Manual Volume 1; Revision 29

1R22 Surveillance Testing

CR 09-56447; SLC Pump B Breather Line Damaged; dated April 1, 2009
Operations Excellence Pre-Job Briefing Checklist, dated July 7, 2009
WO 200289830; (92D) Standby Liquid Control A Pump and Valve Operability Test; dated July 7, 2009
Procedure SVI-C41-T2001-A; Standby Liquid Control A Pump and Valve Operability Test, Revision 12
DWG B 220-0731-00000; Electrical Lighting Panel Details for Lighting Panel 1R71P030; Revision L
Calculation C41-008; Standby Liquid Control System; Revision 97
WO 200261999; HPCS Logic System Functional Test; dated July 20, 2009
Procedure PY-SVI-E22T1192; (24M) HPCS Logic System Functional Test; Revision 7
Procedure SVI-E51-T2001; RCIC Pump and Valve Operability Test; Revision 31
Procedure SVI-P53-T6305; Lower Primary Containment Air Lock (Penetration P305), In Between the Seals Test; Revision 5
Procedure PAP-1120; Leak Testing Program; Revision 5
Procedure FTI-F0031; Volumetrics/FENOC Leak Rate Monitors Testing Instruction; Revision 3
Procedure SVI-E22-T1319; Diesel Generator Start and Load Division 3; Revision 15
Procedure PTI-E22-P0006; Division 3 HPCS Diesel Generator Auxiliary System Monitoring; Revision 9
Procedure SOI-E22B; Division 3 Diesel Generator; Revision 23
CR 09-62090; Procedure Enhancements for SVI-E22-T1192; dated July 20, 2009
CR 09-57416; Division 3 LOOP and LOOP/LOCA Enhancement; dated April 18, 2009
SVI-P53-T6305; Lower Primary Containment Airlock (Penetration P305), In Between the Seals Test; Revision 5
FTI-F0031; Volumetrics/FENOC Leak Rate Monitors Testing Program; Revision 3
PAP-1120; Leak Testing Program; Revision 5

1EP6 Drill Evaluation

Perry Nuclear Power Plant 07-17-2009 ERO Drill Scenario Guide; dated July 14, 2009
07-17-09 ERO Drill Objectives Summary
Drill Critique Summary from TSC; dated July 17, 2009
Drill Critique Summary from Simulator Control Room; dated July 17, 2009
Drill Initial Notification Forms; dated July 17, 2009
Drill Follow-Up Notification Forms; dated July 17, 2009
Drill Event Notification Forms; dated July 17, 2009
CR 09-61989; Simulator Model Limit Exceeded During Eplan Drill; dated July 17, 2009
CR 09-62091; ERO Drill Follow-up Notification to NRC Not Completed Within Required Time; dated July 17, 2009
CR 09-62089; Atmospheric Refrigerant Sampling Issues During Emergency Drill; dated July 17, 2009
CR 09-62088; MARCS Radio Problems During Emergency Drill; dated July 17, 2009
CR 09-62082; Operations Emergency Drill Control Issues; dated July 17, 2009
Procedure PSI-0019; Emergency Action Level (EAL) Bases Document; Revision 12
Perry ERO Drill Guide; dated August 18, 2009

2OS1 Access Control to Radiologically Significant Areas

CR Number 09-55159; Un-posted HRA Found in TB 577 IP Hotwell Drain Line; dated March 11, 2009
CR Number 09-56298; LHRA Entry Without Re-Brief, RWP Violation; dated March 30, 2009
CR Number 09-60846; Area Intermediate Building 574 Found Unposted Radiation Area; dated June 21, 2009
HPI-B0003; Processing of Personnel Dosimetry; Revision 23
NOP-OP-4102; Radiological Posting, Labeling, and Markings; Revision 02
NOP-OP-4107; Radiation Work Permits; Revision 03
PAP-0114; Radiation Protection Program; Revision 14
CR 09-55466; HRA Access Control Event; dated March 15, 2009
CR 09-57546; Dose Alarms Received While Working 1E12F0010; dated April 20, 2009
NOBP-LP-2604; Effective Job Briefs; Revision 0
NOP-OP-4002; Conduct of Radiation Protection; Revision 00
NOP-OP-4101; Access Controls for Radiologically Controlled Areas; Revision 01
CR Number 09-57549; Required RWP Revision Not Promptly Made; dated April 21, 2009
NOBP-LP-2604; Effective Job Briefs; Revision 0
NOP-OP-4002; Conduct of Radiation Protection; Revision 00
NOP-OP-4101; Access Controls for Radiologically Controlled Areas; Revision 01
NOP-OP-4107; Radiation Work Permits; Revision 03
PAP-0114; Radiation Protection Program; Revision 14

2OS2 As-Low-As-Is-Reasonably-Achievable Planning and Controls

NOBP-OM-4009; Outage Scope Identification and Control; Revision 04
NOBP-OP-4109; ALARA Post Outage Report; Revision 00
NOP-OP-4005; ALARA Program; Revision 01
NOP-OP-4202; Declared Pregnant Radiation Workers; Revision Draft A
PAP-0114; Radiation Protection Program; Revision 12
Selected Station Records for Declared Pregnant Workers; Various dates 2009
Station ALARA Committee Meeting Notes; Various dates 2008 and 2009

Radiation Work Permit and Associated ALARA Files; RWP 096035; RFO-12 Alternate Decay Heat Removal Project; Revision 0
Radiation Work Permit and Associated ALARA Files; RWP 096040; RFO-12 Suppression Pool Diving Activities; Revision 0
Radiation Work Permit and Associated ALARA Files; RWP 096057; Pre-Outage Scaffold; Revision 0
RWP Dose Estimate Tracking and Worksheets; Various dates 2008 2009

4OA1 Performance Indicator Verification (71151)

CR Number 09-56439; Fourth Quarter TLD Read Higher Than MG Estimate; dated February 10, 2009
NOBP-LP-4012; NRC Performance Indicators; Revisions 03
Selected Submittals for Perry Performance Indicator Occurrences; Various dates 2008 and 2009
Perry Safety System Functional Failures; July 2008 to June 2009
Perry Unplanned Power Changes Per 7,000 Critical Hours Input; July 2008 to June 2009
Reactor Coolant System Leakage Input; July 2008 to June 2009
LER 2008-003; Inoperable High Pressure Core Spray System Results in Loss of Safety Function; dated July 28, 2008
LER Supplement 2008-003-01; Inoperable High Pressure Core Spray System Results in Loss of Safety Function; dated September 26, 2009

4OA2 Identification and Resolution of Problems

PDF CR List for period January 1, 2009 to March 31, 2009 provided by Perry Licensing via E-Mail from Lloyd Zerr, on September 14, 2009
PDF CR List for period April 1, 2009 to June 30, 2009 provided by Perry Licensing via E-Mail from Lloyd Zerr, on September 14, 2009

4OA3 Followup of Events and Notices of Enforcement Discretion

Chemical Evaluation Request & Permit, Fyrquel EHC Fluid, Revision 6
Material Data Safety Sheet; Fyrquel EHC Fluid; dated August 30, 2004
Control Room Narrative Log, July 23, 2009 and July 24, 2009
PAP-0806; Oil/Chemical Release Contingency Plan; Revision 12
WO 200321593; Bypass Valve Timing Test with Accumulators Isolated; dated April 14, 2009
SVI-C85TO422; Turbine Bypass System Response Time and System Functional Test; Revision 2
CR 09-63681; Loss of Buss EF1A; dated August 25, 2009
LER 2009-002; Diesel Generator CO2 Fire Suppression Control Panel Miswiring Results in an Unanalyzed Condition; dated August 31, 2009
NOP-WM-3001; Work Management PM Process; Revision 7
WO 200297036; Main Supply Brkr from Bus EH11; Brkr EF1; dated August 19, 2009
SOI-R23; 480 Volt Load Centers, Revision 8
SOI-1R10(LV); Plant Electrical Systems; Revision 13
WO 200386812; Troubleshoot Cause of APRM “A” Inoperability; dated September 9, 2009
CR 09-63720; Loss of EF1A Event Crew Performance Critique; dated August 25, 2009
SOI-R22; Metal Clad Switchgear 5-15KV; Revision 23

LIST OF ACRONYMS USED

ALARA	as low as reasonably achievable
APRM	average power range monitor
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	*Code of Federal Regulations*
CO_2	carbon dioxide
CR	condition report
ECC	emergency core cooling
ED	electronic dosimeters
EDG	emergency diesel generator
EHC	electro-hydraulic control
EOP	Emergency Operating Procedure
ESW	emergency service water
FENOC	FirstEnergy Nuclear Operating Company
FSAR	Final Safety Analysis Report
GNPM	General Nuclear Preventative Maintenance
HPCS	high pressure core spray
HVAC	heating, ventilation, and air conditioning
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
LER	Licensee Event Report
LHRA	locked high radiation area
LPCI	low pressure core injection
LPCS	low pressure core spray
MSR	moisture separator reheater
NCV	non-cited violation
NEI	Nuclear Energy Institute
NOP	Nuclear Operating Procedure
NRC	Nuclear Regulatory Commission
OE	Operating Experience
ONI	Off-Normal Instruction
PAP	Perry Administrative Procedure
PI	performance indicator
PMI	Preventative Maintenance Instruction
PS	power supply
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RFBP	reactor feed booster pump
RHR	residual heat removal
RP	radiation protection
RPM	radiation protection manager
RWP	radiation work permit
SDP	Significance Determination Process
SVI	Surveillance Instruction
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report

V	volts
Vdc	volts direct current
WO	work order

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Sincerely,

/RA/

Jamnes L. Cameron, Chief
Branch 6
Division of Reactor Projects

Docket No. 50-440
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Letter to M. Bezilla from J. Cameron dated November 4, 2009.

SUBJECT: PERRY NUCLEAR POWER PLANT NRC INTEGRATED INSPECTION REPORT 05000440/2009004

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