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

July 28, 2011

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 05000440/2011003

Dear Mr. Bezilla:

On June 30, 2011, the U.S. Nuclear Regulatory Commission (NRC) completed a routine inspection at your Perry Nuclear Power Plant. The enclosed report documents the results of this inspection, which were discussed on July 11, 2011, with you and other members of your staff.

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

Based on the results of this inspection, one self-revealed and two NRC-identified findings of very low safety significance were identified. Two of the findings involved a violation of NRC requirements. Additionally, one licensee-identified violation is described in Section 4OA7 of this report. However, because of their very low safety significance, and because the issues were entered into your corrective action program, the NRC is treating the issues as non-cited violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

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

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000440/2011003
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:	05000440/2011003
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, Ohio
Dates:	April 1, 2011, through June 30, 2011
Inspectors:	M. Marshfield, Senior Resident Inspector T. Hartman, Resident Inspector J. Cassidy, Senior Health Physicist D. Jones, Reactor Inspector V. Myers, Health Physicist (NSPDP) R. Orlikowski, Project Engineer M. Phalen, Senior Health Physicist P. Smagacz, Reactor Engineer
Approved by:	Jamnes Cameron, Branch Chief Branch 6 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000440/2011003, 04/01/2011 – 06/30/2011, Perry Nuclear Power Plant; Operability Determinations and Functionality Assessments; Follow-up of Events and Notices of Enforcement Discretion.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three green findings, two 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). The cross-cutting aspects were determined using IMC 0310, "Components Within the Cross-Cutting Areas." Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Initiating Events

Green. A finding of very low safety significance and associated NCV of Technical Specification (TS) 5.4.1 was self-revealed for the licensee's failure to follow plant procedures. The inspectors determined that the licensee failed to follow a procedure which requires verification of expected effects when operating plant components. This failure led to draining approximately 15,000 gallons of suppression pool water which overflowed the Auxiliary Building sump and caused the spread of contamination to various areas of the Auxiliary Building. The licensee entered the issue into their corrective action program. Immediate actions included securing all sources of water to the Auxiliary Building sump and removing water from the Auxiliary Building.

This performance deficiency was determined to be more than minor because it impacted the Human Performance attribute of the Initiating Events Cornerstone, and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding is of very low safety significance because it did not increase the likelihood of a loss of reactor coolant system inventory, degrade the licensee's ability to terminate a leak path or add inventory, or degrade the licensee's ability to recover decay heat removal. The finding was associated with a cross-cutting aspect in the Resources component of the Human Performance cross-cutting area per IMC 0310 [H.2(c)], because the licensee did not provide complete, accurate and up-to-date procedures. Specifically, the procedure to test the residual heat removal waterleg pump did not address the potential to drain the suppression pool to the Auxiliary Building sump. (Section 4OA3.1)

Cornerstone: Mitigating Systems

Green. The inspectors identified a finding of very low safety significance and associated NCV of TS 5.4.1.a, for failure to establish a procedure to remove power from the shutdown cooling isolation (SDC) valves while shutdown cooling was in operation during a plant refueling outage. The inspectors determined that the licensee performed an

activity that affected quality without a proper procedure in place. The licensee entered the issue into their corrective action program.

This performance deficiency was determined to be more than minor because it impacted the Procedure Quality attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). This finding is of very low safety significance because the risk significance was evaluated to have a delta core damage frequency of less than E-6/yr and a delta large early release frequency of less than E-7/yr. This finding was associated with a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area per IMC 0310 [H.4(b)] because the licensee did not effectively communicate expectations regarding procedural compliance and personnel following procedures. Specifically, the operators did not question operating safety-related plant equipment without appropriate procedural guidance. (Section 1R15)

Green. The inspectors identified a finding of very low safety significance for failure to follow TS Limiting Condition for Operations 3.0.2 bases. The inspectors determined that the licensee rendered safety-related plant equipment inoperable and entered TS 3.6.1.3 Condition A for operational convenience. The licensee entered the issue into their corrective action program.

This performance deficiency was determined to be more than minor because it impacted the Configuration Control attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). This finding is of very low safety significance because it did not increase the likelihood that a loss of decay heat removal, reactor coolant system inventory, or offsite power would occur and did not degrade the ability to terminate a leak path, recover decay heat removal once it is lost, or establish an alternate core cooling path if decay heat removal could be re-established. This finding was associated with a cross-cutting aspect in the Decision Making component of the Human Performance cross-cutting area per IMC 0310 [H.1(b)] because the licensee did not use conservative assumptions to ensure the proposed action was safe. Specifically, the licensee chose to disable automatic protective features of a plant system while performing “high-risk” activities. (Section 1R15)

B. Licensee-Identified Violations

One violation of very low safety significance identified by the licensee was reviewed by the inspectors. Corrective actions taken or planned by the licensee have been entered into the licensee’s corrective action program. This violation and corrective action tracking number 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 April 18, 2011, at 0001 hours the plant was shut down for Refueling Outage (RFO) 13. On June 5, 2011, at 0806 hours the plant was placed in startup mode and achieved criticality at 1006 hours on the same day. On June 7, 2011, at 0035 hours the plant generator was synchronized to the grid ending the RFO. On June 9, 2011, at 1023 hours the plant separated from the grid and reactor power was reduced to 8 percent to support plant repairs on a broken steam line test connection. On June 12, 2011, at 0022 hours the plant synchronized to the grid and began power ascension. The plant achieved 100 percent power on June 14, 2011, at 0026 hours. On June 21, 2011, at 0605 hours, power was reduced to 75 percent to investigate the closure of Turbine Control Valve 4. On June 22, 2011, repairs were completed and the plant returned to 100 percent power at 0732 hours. The plant operated at full power with only minor variations for the remainder of the quarter.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 External Flooding

a. Inspection Scope

The inspectors evaluated the design, material condition, and procedures for coping with the design basis probable maximum flood. The evaluation included a review to check for deviations from the descriptions provided in the Updated Safety Analysis Report (USAR) for features intended to mitigate the potential for flooding from external factors. As part of this evaluation, the inspectors checked for obstructions that could prevent draining, checked that the roofs did not contain obvious loose items that could clog drains in the event of heavy precipitation, and determined that barriers required to mitigate the flood were in place and operable. Additionally, the inspectors performed a walkdown of the protected area to identify any modification to the site which would inhibit site drainage during a probable maximum precipitation event or allow water ingress past a barrier. The inspectors also reviewed the Off-Normal Instruction (ONI) for mitigating the design basis flood to ensure it could be implemented as written.

This inspection constituted one external flooding sample as defined in Inspection Procedure (IP) 71111.01-05.

b. Findings

No findings were identified.

.2 Summer Seasonal Readiness Preparations

a. Inspection Scope

The inspectors performed a review of the licensee's preparations for summer weather for selected systems, including conditions that could lead to an extended drought.

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 USAR and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant-specific procedures. Specific documents reviewed during this inspection are listed in the Attachment. 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. The inspectors' reviews focused specifically on the turbine building chilled water system.

This inspection constituted one sample for seasonal adverse weather as defined in IP 71111.01-05.

b. Findings

No findings were identified.

.3 Readiness of Offsite and Alternate AC Power Systems

a. Inspection Scope

The inspectors verified that plant features and procedures for operation and continued availability of offsite and alternate alternating current (AC) power systems during adverse weather were appropriate. The inspectors reviewed the licensee's procedures affecting these areas and the communications protocols between the transmission system operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Examples of aspects considered in the inspectors' review included:

- coordination between the TSO and the plant during off-normal or emergency events;
- explanations for the events;
- estimates of when the offsite power system would be returned to a normal state; and
- notifications from the TSO to the plant when the offsite power system was returned to normal.

The inspectors also verified that plant procedures addressed measures to monitor and maintain availability and reliability of both the offsite AC power system and the onsite alternate AC power system prior to or during adverse weather conditions. Specifically, the inspectors verified that the procedures addressed the following:

- actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant would not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;
- compensatory actions identified to be performed if it would not be possible to predict the post-trip voltage at the plant for the current grid conditions;
- re-assessment of plant risk based on maintenance activities which could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power was challenged.

Documents reviewed are listed in the Attachment to this report. The inspectors also reviewed 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.

This inspection constituted one sample for readiness of offsite and alternate AC power systems as defined in IP 71111.01-05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Partial System Walkdowns

a. Inspection Scope

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

- low-pressure core spray (LPCS) on April 22, 2011;
- 'A' motor control center, switchgear and miscellaneous electrical equipment heating ventilation and air conditioning (HVAC) system during planned maintenance on 'B' motor control center, switchgear and miscellaneous electrical equipment HVAC system on June 23, 2011; and
- standby liquid control system (SBLC) on June 27, 2011.

The inspectors selected these systems based on their risk-significance relative to the Reactor Safety Cornerstone at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, USAR, Technical Specification (TS) requirements, outstanding work orders (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 inspections constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings were identified.

.2 Complete System Walkdown

a. Inspection Scope

On June 20, 2011, 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 to this report.

This inspection constituted one complete system walkdown sample as defined in IP 71111.04-05.

b. Findings

No findings were identified.

.3 System Walkdown Associated with Temporary Instruction 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems."

a. Inspection Scope and Documentation

On May 18, 2011, the inspectors conducted a walkdown of the LPCS system in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (Temporary Instruction (TI) 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspector's independent walkdown (TI 2515/177, Section 04.02.c.3).

In addition, the inspectors verified that the licensee had isometric drawings that describe the LPCS system configurations and had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors verified the following related to the isometric drawings:

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

The inspectors verified that Piping and Instrumentation Diagrams (P&IDs) accurately described the subject systems, that they were up to date with respect to recent hardware changes, and any discrepancies between as-built configurations, the isometric drawings, and the P&IDs were documented and entered into the CAP for resolution (TI 2515/177, Section 04.02.b).

Documents reviewed are listed in the Attachment to this report.

This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later inspection report (IR).

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

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

- Fire Zones 0CC-1a,b,c (Control Complex 574');
- Fire Zone TB-577 (Turbine Building 577');
- Fire Zone 0IB-3 (Intermediate Building 620');
- Fire Zone 1RB-1C-1c (Drywell); and
- Fire Zone 0FH-3 (Fuel Handling Building 620').

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 five quarterly samples for fire protection as defined in IP 71111.05-05.

b. Findings

No findings were identified.

1R06 Flood Protection Measures (71111.06)

Internal Flooding

a. Inspection Scope

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

- Auxiliary Building 599' level;
- Auxiliary Building 574' level; and
- emergency core cooling system (ECCS) rooms.

This inspection constituted one internal flooding sample as defined in IP 71111.06-05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

a. Inspection Scope

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

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

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08G)

From April 25 through April 29, 2011, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system (RCS), risk-significant piping and components and containment systems.

The ISIs described in Sections 1R08.1 and 1R08.2 below constituted one inspection sample as defined in IP 71111.08-05.

.2 Piping Systems ISI

a. Inspection Scope

The inspectors observed the following non-destructive examinations mandated by the American Society of Mechanical Engineers (ASME) Section XI Code to evaluate compliance with the ASME Code Section XI and Section V requirements and if any indications and defects were detected, to determine if these were dispositioned in accordance with the ASME Code or an NRC-approved alternative requirement.

- Ultrasonic Examination (UT) of the Top Head Meridional Weld @ 255 AZ, Report No. UT-11-E016;
- Ultrasonic Examination of the Top Head Meridional Weld @ 75 AZ, Report No. UT-11-E015; and
- Visual Examination of the Steam Dryer Hold Down Bracket/Vessel Weld, Report No. 1042-11-059.

The inspectors reviewed the following examinations completed during the previous outage with relevant/recordable conditions/indications accepted for continued service to determine if acceptance was in accordance with the ASME Code Section XI or an NRC-approved alternative.

- Evaluation of Reactor Vessel Residual Heat Removal Nozzle N6C to Safe-End Weld (1B13-N6C-KB), CR 09-56393.

The inspectors reviewed the following pressure boundary weld completed for a risk-significant system since the beginning of the last RFO to determine if the licensee applied the pre-service non-destructive examinations and acceptance criteria required by the ASME Code Section XI. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure was qualified in accordance with the requirements of Construction Code and the ASME Code Section IX.

- Removal and Reinstallation of Feedwater Check Valve 1B21-F032B Test Connection, WO No. 200262683.

b. Findings

No findings were identified.

.3 Identification and Resolution of Problems

a. Inspection Scope

The inspectors performed a review of ISI-related problems entered into the licensee's CAP and conducted interviews with licensee staff to determine if:

- the licensee had established an appropriate threshold for identifying ISI-related problems;
- the licensee had performed a root cause (if applicable) and taken appropriate corrective actions; and
- the licensee had evaluated operating experience and industry generic issues related to ISI and pressure boundary integrity.

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements. The corrective action documents reviewed by the inspectors are listed in the Attachment to this report.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

On May 27, 2011, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator just-in-time training to support start-up from RFO 13. The inspectors verified that operator performance was adequate, evaluators were

identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12Q)

a. Inspection Scope

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

- division 2 emergency diesel generator (EDG); and
- upper containment airlock

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

b. Findings

No findings were identified.

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

a. Inspection Scope

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

- decay heat removal (DHR) equipment during shutdown defense-in-depth phase 1 and 2 of RFO 13;
- shutdown risk with L-H-1A, “interbus transformer,” out of service;
- control rod drive mechanism (CRDM) changeouts and response to dropped CRDM;
- updated defense-in-depth for second half of RFO 13;
- high-pressure signal jumper installation to prevent loss of RHR cooling in Modes 4 and 5; and
- tagging of source range monitor (SRM) 'D' while SRM 'C' was out-of-service during Mode 5 operations.

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 six samples as defined in IP 71111.13-05.

b. Findings

No findings were identified.

1R15 Operability Evaluations and Functionality Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- shutdown cooling isolation valves power removal;
- reactor feedwater pump 'A' discharge check valve;
- Residual Heat Removal operability during instrumentation and control (I&C) maintenance; and
- continued operation with recirculation pump 'B' seal degradation.

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

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

b. Findings

.1 Failure to Establish a Procedure to Operate Safety-Related Equipment

Introduction: A finding of very low safety significance (Green) and associated NCV of TS 5.4.1.a, was identified by the inspectors for failure to establish a procedure to place the RHR system in a condition which rendered required plant equipment inoperable. Specifically, the licensee removed power from the shutdown cooling (SDC) isolation valves while SDC was in operation without proper plant procedures, which prevented the isolation valves from automatically closing in the event a loss-of-coolant accident (LOCA) occurred.

Description: On April 20, 2011, during a review of operations standing orders, the inspectors noted a standing order that discussed actions to minimize the potential of an inadvertent isolation of SDC. The order noted several valves that would have their power removed to prevent inadvertent closure and subsequent loss of SDC. The most notable of these valves were the SDC isolation valves that are designed to close on lowering reactor pressure vessel (RPV) water level to minimize the loss of RPV inventory during a LOCA.

The inspectors determined that there was no approved procedure related to the control of this evolution. Instead, the licensee gave guidance through an Operations Standing Order. This is contrary to TS 5.4.1.a, which requires that written procedures be established, implemented, and maintained covering activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978, which includes operation of safety-related systems. The Operations Standing Order was generated and approved by the Superintendent of Plant Operations. No other official reviews were required or documented. Also, because there was no procedure, there were no specific actions to direct recovery of the valves if they were called upon to support their safety-related function which consequently rendered the safety function unavailable.

A review of narrative log entries identified nine instances within a 6-day period when power was removed from these required isolation valves. The total combined time the valves were inoperable was approximately 11.5 hours. The licensee entered the associated TS LCO for these valves being inoperable. No instances where the LCO-required action times were exceeded were identified.

Analysis: The inspectors determined that the licensee's failure to establish a procedure to realign and render plant equipment inoperable constituted a performance deficiency. Specifically, the licensee removed power from the SDC isolation valves, rendering them unable to perform their design function, without an approved plant procedure. The inspectors evaluated the performance deficiency in accordance with IMC 0612, Appendix B, "Issue Screening." This performance deficiency was not similar to any of the examples in IMC 0612, Appendix E, "Examples of Minor Issues," but was characterized as more than minor because it impacted the Procedure Quality attribute of the Mitigating Systems cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage).

The inspectors determined the finding could be evaluated using the SDP in accordance with Inspection Manual Chaper (IMC) 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of Findings," Table 3b for the Mitigating Systems Cornerstone. This directed the inspectors to IMC 0609, Appendix G "Shutdown Operations Significance Determination Process."

The Senior Reactor Analysts (SRAs) determined that IMC 0609, Appendix G directed the use of Attachment 1, "Shutdown Operations Significance Determination Process, Phase 1 Operational Checklists for Both PWRs and BWRs." The SRAs used Checklists 6 and 7 contained in Appendix G, Attachment 1, and determined that the finding required a Phase 2 analysis since the finding degraded the ability to terminate a leak path or add RCS inventory when needed (Section II.A of Checklists 6 and 7).

The SRAs performed a modified Phase 2 assessment using Appendix G, Attachment 3, "Phase 2 Significance Determination Process Template for BWR During Shutdown." The SRAs determined this to be a "condition finding" since it involved the degradation of the capability to mitigate a loss of RCS inventory event if an event were to occur. Based on plant data, two plant operational states (POSs) were determined to apply during the exposure time. One POS was determined to be "POS 1" (vessel head on and RCS closed) with an exposure time of less than 5 hours. The second POS was determined to be "POS 3" (vessel head off with RPV water level equal to or greater than the minimum level required for movement of irradiated fuel assemblies within the RPV as defined by

TS) with an exposure time of less than 7 hours. Based on the exposure times, the initiating event likelihood (IEL) for loss of inventory (LOI) was determined to be “5” for each POS.

Using Appendix G, Attachment 3, Worksheet 1, "SDP for a BWR Plant - Loss of Inventory in POS 1 (Head on),” the analysts evaluated the risk significance while in POS 1 by evaluating the remaining mitigating capability credit to reflect equipment availability and operator credit to complete tasks prior to core damage. The most significant core damage sequence in POS 1 involved a loss of RCS inventory with failure to isolate the LOI and failure of the operator to open an RCS vent path (i.e., a safety relief valve). This sequence had a risk-significance of “7” or a delta core damage frequency of about 3.3E-7/yr.

Using Appendix G, Attachment 3, Worksheet 3, "SDP for a BWR Plant - Loss of Inventory in POS 3 (Cavity Flooded),” the analysts evaluated the risk significance while in POS 3 by evaluating the remaining mitigating capability credit to reflect equipment availability and operator credit to complete tasks prior to core damage. The most significant core damage sequence in POS 3 involved a loss of RCS inventory with failure to isolate the LOI and failure of the operator to reconfigure RHR to ECCS injection. This sequence had a risk-significance of “8” or a delta core damage frequency of about 3.3E-8/yr.

The total risk significance is the sum of the risk contributions while in POS 1 and the risk contribution while in POS 3. The risk significance was thus evaluated to be a delta core damage frequency of approximately 3.6E-7/yr.

Since the delta core damage frequency was greater than E-7/yr, the potential risk contribution for this finding from large early release frequency (LERF) was evaluated using the guidance of IMC 0609 Appendix H, “Containment Integrity Significance Determination Process.” Using Table 5.4, “Phase 2 Assessment Factors – Type A Findings at Shutdown,” of IMC 0609 Appendix H, the delta LERF was evaluated to be less than E-7/yr.

Based on the Phase 2 analysis, the SRAs determined that the finding was of very low safety-significance (Green).

This finding has a cross-cutting aspect in the Work Practices component of the Human Performance cross-cutting area per IMC 0310 [H.4(b)] because the licensee did not effectively communicate expectations regarding procedural compliance and personnel following procedures. Specifically, the operators did not question operating safety-related plant equipment without appropriate procedural guidance.

Enforcement: A requirement of TS 5.4.1.a is that written procedures be established, implemented, and maintained covering activities recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978, which includes operation of safety-related systems. Contrary to the above, between April 19, and April 25, 2011, the licensee manipulated safety-related plant equipment without a governing plant procedure. Specifically, the licensee removed power from the SDC isolation valves, in an open position, while the SDC system was in service. This action rendered the SDC isolation valves inoperable and caused an unnecessary entry into a TS action. Because this violation was of very low safety significance and it was entered into the licensee’s CAP as CR 11-94572, this violation is being treated as an NCV, consistent with

Section 2.3.2 of the NRC Enforcement Policy. (NCV 05000440/2011003-01, Failure to Establish a Procedure to Operate Safety-Related Equipment)

.2 Failure to Follow Technical Specification Bases

Introduction: A finding (FIN) of very low safety significance (Green) was identified by the inspectors for failure to follow the TS bases associated with LCO 3.0.2. Specifically, the licensee rendered safety-related plant equipment inoperable and entered TS 3.6.1.3 Condition A for operational convenience.

Description: On April 20, 2011, during a review of narrative logs, the inspectors identified several log entries where the licensee entered TS LCO 3.6.1.3 Condition A to support plant activities. Discussions with the licensee revealed these entries were made to minimize the potential of losing shutdown cooling.

When a power plant is recently shut down, there is a short amount of time before undesired boiling begins to occur in the core (e.g. less than 2 hours) and losing DHR/SDC is a significant concern. An actuation of the containment isolation logic is an event that would result in a loss of SDC. This is because the SDC suction valves are closed during an isolation and therefore SDC flow is stopped. The licensee addressed this concern by removing power from the SDC isolation valves while performing activities that had a "high risk" of actuating the isolation logic. However, industry experience shows one of the likely plant conditions leading to a loss of reactor coolant inventory is while realigning plant systems during a shutdown.

Technical Specification LCO 3.0.2 bases states, in part, "The reasons for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventative maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience. Alternatives that would not result in redundant equipment being inoperable should be used instead." Contrary to the above, the licensee entered the ACTIONS to minimize the potential for an inadvertent isolation actuation and subsequent loss of SDC. Even though the entries were in support of surveillance and maintenance activities, these activities did not require the removal of these valves, nor were the valves directly impacted by the activities. The licensee used the valves to compensate for physical plant design deficiencies, scheduling issues, and human performance concerns. Intentionally entering this ACTION did compromise safety and resulted in redundant equipment being inoperable.

A subsequent review of narrative log entries identified nine instances within a 6-day period where the licensee entered TS 3.6.1.3 Condition A for the SDC isolation valves. The total time the LCO was not met was approximately 11.5 hours. No instance where the LCO-Required Action times were exceeded was identified.

Analysis: The inspectors determined that the licensee's entry into LCO 3.6.1.3 Condition A was for operational convenience and constituted a performance deficiency. Specifically, the licensee rendered the SDC isolation valves inoperable and entered the associated LCO Condition and Required Actions to compensate for physical plant design deficiencies, scheduling issues, and human performance concerns. The inspectors evaluated the performance deficiency in accordance with IMC 0612,

Appendix B, “Issue Screening.” This performance deficiency was not similar to any of the examples in IMC 0612, Appendix E, “Examples of Minor Issues," but was characterized as more than minor because it impacted the Configuration Control attribute of the Mitigating Systems Cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage).

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 3b for the Mitigating Systems Cornerstone. This directed the inspectors to IMC 0609, Appendix G “Shutdown Operations Significance Determination Process.” The inspectors determined that IMC 0609, Appendix G directed one to IMC 0609, Appendix G, Attachment 1, “Shutdown Operations Significance Determination Process, Phase 1 Operational Checklists for Both PWRs and BWRs.” The inspectors used Checklists 6 and 7 contained in Appendix G, Attachment 1, and determined the finding was of very low safety significance (Green) because it did not increase the likelihood that a loss of DHR, a loss of RCS inventory, or a loss of offsite power (LOOP) would occur, nor degrade the ability to terminate a leak path, to recover DHR once it is lost, or to establish an alternate core cooling path if DHR cannot be re-established.

This finding has a cross-cutting aspect in the Decision Making component of the Human Performance cross-cutting area per IMC 0310 [H.1(b)] because the licensee did not use conservative assumptions to ensure the proposed action was safe. Specifically, the licensee chose to disable automatic protective features of a plant system while performing “high-risk” activities.

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. (FIN 05000440/2011003-02 Failure to Follow TS Bases)

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed the modification 10-0266-001; “Upgrade the Existing 480-Volt Motor Control Center (MCC) Automatic Transfer Switch and Relay in MCC FC108.”

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

impact overall plant performance. Documents reviewed in the course of this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- SRM 'C' testing after replacement activities on May 3, 2011;
- division 3 post-electrical relay replacement testing on May 4, 2011;
- division 2 EDG post-outage maintenance testing on May 17, 2011;
- RHR 'B' and 'C' waterleg pump replacement testing on May 19,2011;
- division 2 ATWS-RPT logic system functional test on May 28, 2011;
- reactor core isolation cooling minimum flow valve on June 6, 2007;
- recirculation flow control valve 'B' linear variable transducer replacement on June 10, 2011; and
- unit 2 plant vent noble gas radiation monitor on June 23, 2011.

These activities were selected based upon the structure, system, or component's ability to impact risk. The inspectors evaluated these activities for the following (as applicable): the effect of testing on the plant had been adequately addressed; testing was adequate for the maintenance performed; acceptance criteria were clear and demonstrated operational readiness; test instrumentation was appropriate; tests were performed as written in accordance with properly reviewed and approved procedures; equipment was returned to its operational status following testing (temporary modifications or jumpers required for test performance were properly removed after test completion); and test documentation was properly evaluated. The inspectors evaluated the activities against TS, the USAR, 10 CFR Part 50 requirements, licensee procedures, and various NRC generic communications to ensure that the test results adequately ensured that the equipment met the licensing basis and design requirements. In addition, the inspectors reviewed corrective action documents associated with PM 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 eight PM testing samples as defined in IP 71111.19-05.

b. Findings

No findings were identified.

1R20 Refueling Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for RFO 13, conducted April 18 through June 7, 2011, to confirm that the licensee had appropriately considered risk, industry experience, and previous site-specific problems in developing and implementing a plan that assured maintenance of defense-in-depth. During the RFO, the inspectors observed portions of the shutdown and cooldown processes and monitored licensee controls over the outage activities listed below. Documents reviewed during the inspection are listed in the Attachment to this report.

- Licensee configuration management, including maintenance of defense-in-depth commensurate with the OSP for key safety functions and compliance with the applicable TS when taking equipment out of service.
- Implementation of clearance activities and confirmation that tags were properly hung and equipment appropriately configured to safely support the work or testing.
- Installation and configuration of reactor coolant pressure, level, and temperature instruments to provide accurate indication, accounting for instrument error.
- Controls over the status and configuration of electrical systems to ensure that TS and OSP requirements were met, and controls over switchyard activities.
- Monitoring of DHR processes, systems, and components.
- Controls to ensure that outage work was not impacting the ability of the operators to operate the spent fuel pool cooling system.
- Reactor water inventory controls including flow paths, configurations, and alternative means for inventory addition, and controls to prevent inventory loss.
- Controls over activities that could affect reactivity.
- Maintenance of secondary containment as required by TS.
- Licensee fatigue management, as required by 10 CFR 26, Subpart I.
- Refueling activities, including fuel handling. No fuelsipping to detect fuel assembly leakage was conducted during RFO 13.
- Startup and ascension to full power operation, tracking of startup prerequisites, walkdown of the drywell (primary containment) to verify that debris had not been left which could block emergency core cooling system suction strainers, and reactor physics testing.
- Licensee identification and resolution of problems related to RFO activities.

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

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the test results for the following activities to determine whether risk-significant systems and equipment were capable of performing their intended safety

function and to verify testing was conducted in accordance with applicable procedural and TS requirements:

- emergency closed cooling (ECC) 'A' pump and valve 24-month on April 4, 2011 (inservice testing);
- end-of-cycle, recirculation pump trip (EOC-RPT) arc suppression time response testing on April 17, 2011 (routine testing);
- refuel bridge and mode selector switch interlock on April 26, 2011 (routine testing);
- division 2 standby diesel generator LOOP and LOCA testing on May 16, 2011,(routine testing);
- licensee actions to analyze drywell unidentified leakage on startup from RFO 13 on June 6 and 7, 2011, (RCS leakage); and
- upper and lower primary containment air lock in between the seals testing on June 27, 2011 (routine testing).

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

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges, and the calibration frequency 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 inservice 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;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

The inspection activities supplement those documented in IR 05000440/2011002 (ADAMS Accession Number ML111180447) and constitute a partial sample as defined in IP 71124.01-05.

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

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

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

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

- RFO-13 ECCS Valve Repair Activities;
- RFO-13 Refueling Activities;
- RFO-13 SRM-C Cable Reinsertion; and
- RFO-13 SRM Cable/Detector Removal, Transport, Storage in the FHN Pool and Support Work.

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

- identification of hot particles;
- the presence of alpha emitters;

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

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

b. Findings

A finding was identified when the licensee authorized the removal of SRM 'C' from the reactor vessel, without an evaluation that appropriately identified and assessed the radiological hazards of the work activity. This event was fully evaluated as part of a special inspection and detailed in IR 05000440/2011013 (ADAMS Accession Number ML11187A121).

.2 Instructions to Workers (02.03)

a. Inspection Scope

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

- RWP 116030; RFO-13 ECCS Valve Repair Activities;
- RWP 116019; RFO-13 Refueling Activities;
- RWP 116037; RFO-13 SRM-C Cable Reinsertion; and
- RWP 116038; RFO-13 SRM Cable/Detector Removal, Transport, Storage in the FHN Pool and Support Work.

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

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

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

b. Findings

A finding was identified when the licensee authorized the removal of SRM 'C' from the reactor vessel, without an evaluation that appropriately identified and assessed the radiological hazards of the work activity. This event was fully evaluated as part of a Special Inspection and detailed in IR 05000440/2011013.

.3 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

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

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

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

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

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

- RWP 116030; RFO-13 ECCS Valve Repair Activities;
- RWP 116019; RFO-13 Refueling Activities;
- RWP 116037; RFO-13 SRM-C Cable Reinsertion; and
- RWP 116038; RFO-13 SRM Cable/Detector Removal, Transport, Storage in the FHN Pool and Support Work.

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

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and

physical controls) were in place to preclude inadvertent removal of these materials from the pool.

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

b. Findings

A finding was identified when the licensee authorized the removal of source range monitor 'C' (SRM-C) from the reactor vessel, without an evaluation that appropriately identified and assessed the radiological hazards of the work activity. This event was fully evaluated as part of a Special Inspection and detailed in IR 05000440/2011013.

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

a. Inspection Scope

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

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

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

b. Findings

No findings were identified.

.5 Radiation Worker Performance (02.07)

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.6 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.7 Problem Identification and Resolution (02.09)

a. Inspection Scope

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

b. Findings

No findings were identified.

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

The inspection activities supplement those documented in IR 05000440/2010003 (ADAMS Accession Number ML102170045), and constitute a partial sample as defined in IP 71124.02-05.

Radiation Worker Performance (02.05)

a. Inspection Scope

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

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

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

4OA1 Performance Indicator Verification (71151)

.1 Safety System Functional Failures

a. Inspection Scope

The inspectors sampled licensee submittals for the Safety System Functional Failures performance indicator (PI) for the period from second quarter 2010 through the first quarter 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the Nuclear Energy Institute (NEI) Document 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6, 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 IRs for the period second quarter 2010 through the first quarter 2011 to validate the accuracy of the submittals. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one sample for safety system functional failures as defined in IP 71151-05.

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index - Emergency Alternating Current Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency AC Power System PI for the period from the second quarter 2010 through the first quarter 2011. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in NEI Document 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, was used. The inspectors reviewed the licensee's operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC Integrated IRs for the period of the second quarter 2010 through the first quarter 2011 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one sample for MSPI emergency AC power system PI as defined in IP 71151-05.

b. Findings

No findings were identified.

.3 Mitigating Systems Performance Index – High-Pressure Injection System

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI – High-Pressure Injection System PI for the period from the second quarter 2010 through the first quarter 2011. 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 6, was used. The inspectors reviewed the licensee's operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC Integrated IRs for the period of the second quarter 2010 through the first quarter 2011 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one sample for MSPI high-pressure injection system PI as defined in IP 71151-05.

b. Findings

No findings were identified.

4OA2 Problem Identification and Resolution (71152)

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

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Selected Issue Follow-Up Inspection Associated with Temporary Instruction 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems": Air Observed During Residual Heat Removal 'C' Low-Pressure Core Injection Valve Lineup Verification and System Venting

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized a corrective action item documenting an air void observed in the RHR 'C' low pressure core injection (LPCI) piping. Condition Report 11-90863 documents that during the performance of the RHR 'C' LPCI valve lineup and system vent monthly surveillance on March 11, 2011, after 1.5 minutes of solid stream venting, an air/water mixture was observed at the high point of the RHR 'C' line. Perry engineering determined that the air volume was approximately 0.1675 cubic feet. Perry's 9-month response to Generic Letter (GL) 2008-01 indicates the LPCI train could tolerate an air volume of 1.0 cubic feet without the discharge piping relief valves lifting during a hydraulic transient. A conservative calculation was performed showing that in a 10-foot length of 1-inch pipe, the void would be 0.1703 cubic feet with the waterleg pump shutdown. No other voiding issues were identified by the inspectors.

The inspectors verified that the selected CAP entry acceptably addressed the areas of concern associated with the scope of GL 2008-01, "Managing Gas Accumulation In Emergency Core Cooling, Decay Heat Removal, And Containment Spray Systems" (TI 2515/177, Section 04.01).

This review constituted one in-depth problem identification and resolution sample as defined in IP 71152-05. In addition, this inspection effort counts towards the completion of TI 2515/177 which will be closed in a later IR.

b. Findings

No findings were identified.

.4 Selected Issue Follow-Up Inspection: Review of Root Cause Analysis Report: Preparations of Online Work Windows Has Not Met Work Management Expectations or Milestones

a. Inspection Scope

The inspectors reviewed Perry Plant's "Root Cause Analysis Report: Preparations of Online Work Windows Has Not Met Work Management Expectations or Milestones" as part of a review of the station's performance relative to work planning issues identified in October 2010. Specifically, CR 10-85080 identified that "preparations for the recent Divisional Outages has had less than acceptable adherence with the Work Management process expectations and compliance with the schedule milestones."

In 2010 Perry had five findings within the cross-cutting area component of Work Control. The inspectors chose this root cause analysis report to review Perry's evaluation of their work planning and control issues. The root and contributing causes were reviewed along with the corrective actions implemented to address those causes.

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

b. Assessment and Observations

There was one root cause identified: "Less than adequate management involvement and willingness to develop a strategy to match available resources to existing workload demands." To address this root cause, two prevent recurrence (PR) actions were implemented:

- Prevent Recurrence Action 1 was to "establish a strategy and an action plan to manage existing and future workload demands with the manhour resources provided."

- Prevent Recurrence Action 2 was to "eliminate the excess workload on Shop/Work Coordinators and ensure they perform their required Work Management Duties. Provide sufficient depth and backup for this position. Reduce Training Coordinator duties currently assigned to the Shop/Work Coordinators. (Note: Training Coordinator responsibilities may be returned to the Shop/Work Coordinators at a future date when WM performance issues and Maintenance Training program status are in a position to support this change.)"

On March 15, 2011, CR 11-91031 was initiated by Perry's Nuclear Oversight (NOS) group after reviewing the root cause and its associated corrective actions. This CR documented several issues identified during NOS's review. Two of the issues were related to the two PR actions identified in the root cause analysis. The NOS group stated that "actions to create a team to create further actions to present to CARB [Corrective Action Review Board] are not effective at addressing the cause identified." Also, NOS stated that "actions to prevent recurrence should not have a caveat to remove the actions that have been identified to prevent recurrence."

To address NOS's concerns, a corrective action item was added to CR 10-85080 to evaluate these concerns. With regards to PR Action 2, the evaluation determined that:

- "The caveat in this corrective action was included based on discussions with the CA owner, the Maintenance Director. The limitations of the caveat are appropriate for this action. As long as the Shop Coordinators can still successfully perform their WM [Work Management] responsibilities, having additional Training Coordinator duties will not negate the desired outcome of this corrective action."

With regards to PR Action 1, the evaluation determined that:

- "In order to adequately and accurately address the various corrective actions, having a team of responsible experts develop action plans and strategies to address the issue has been an accepted practice at Perry. The results and output of the teams are then further reviewed by the CARB to ensure that they address the identified causes. In the end, the final product is much better and more robust than one that the root cause team could create."

The final conclusion of the evaluation of NOS's concerns stated that "no changes to the root cause evaluation as documented in CREST are needed as a result of this corrective action."

After reviewing the root cause analysis report, the inspectors had several observations of Perry's root cause evaluation and its associated corrective actions. A PR Action is defined by Perry Nuclear Operating Procedure (NOP)-LP-2001, "Corrective Action Program," as an "action type implemented with the intent to preclude repetition of the deficiency. Preventive Actions are required for Root Causes." NOP-LP-2001 further states that "because of the significance of regulatory or safety consequences associated with significant conditions, PR actions shall be generated to preclude repetition of problems identified as significant conditions. This is required by 10 CFR 50, Appendix B, Criterion XVI."

The inspectors determined that if Training Coordinator duties are returned to the Shop/Work Coordinators, which is allowable by the approved actions and documentation in CR 10-85080, then PR Action 1 may no longer be a PR Action that would prevent recurrence. Further follow-up is recommended by NRC inspectors at a later date to determine if Training Coordinator duties are in fact returned to the Shop/Work Coordinators and what impact that may have on the root cause identified in CR 10-85080.

Perry Procedure NOBP-LP-2011, "FENOC Cause Analysis," step 4.1.1 states that:

- "the following general requirements apply to all condition reports that require a cause evaluation: [10] Document the corrective action plan, including corrective actions taken or needed to restore the condition to acceptable standards, to address all causes (apparent, root and/or contributing) and to address other conditions identified in the cause evaluation."

The NRC inspectors determined that PR Action 1, which recommended a team comprised of Work Management, Maintenance, Operations, and Engineering personnel to develop an action plan and additional corrective actions to implement the plan does not explicitly follow the requirements of procedure NOBP-LP-2011 in that not all of the corrective actions are documented within the PR Action item. Preventive Recurrence Action 1 does document a corrective action to develop and document an action plan, but it is not explicit in what the action plan should require in order to address the root cause identified in CR 10-85080. At the time this inspection was completed, the action plan had been developed but several of the actions had not been completed.

Finally, the inspectors noted that NOS's evaluation of CR 10-85080 and the root cause analysis report identified several issues that were documented in CR 11-91031 on March 15, 2011. These issues and concerns were addressed in corrective action item 26 of CR 10-85080, which was completed on April 6, 2011. The inspectors noted that ultimately no changes were made to the root cause or its corrective actions as a result of NOS's evaluation. Additionally, PR Action 1 was implemented on March 24, 2011 and PR Action 2 was implemented on March 14, 2011, both of which occurred before NOS's issues were addressed in CR 10-85080. The inspectors could not determine if the fact that both PR Action items were implemented prior to addressing NOS's concerns influenced that station's decision to not make any changes to the root cause analysis or it's PR Actions.

c. Findings

No findings were identified.

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

.1 Source Range Monitor 'C' Removal

a. Inspection Scope

The inspectors observed and reviewed the licensee's response to unexpectedly high dose rates encountered while attempting to remove SRM 'C' on April 21, 2011. While attempting to remove a portion of the activated SRM's cable by hand, all workers in the vicinity received dose rate alarms and evacuated the immediate area. Due to the high dose rates observed, the NRC formed a special inspection team (SIT) to respond to this event. Results of this inspection can be found in IR 05000440/2011013, (ADAMS Accession Number ML11187A121). Documents reviewed in this inspection are listed in the Attachment.

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

b. Findings

No findings were identified.

.2 Auxiliary Building Sump Overflow

a. Inspection Scope

The inspectors reviewed the licensee's response to the May 19, 2011, draining of the suppression pool to the Auxiliary Building sump which led to unexpected high sump level alarms in reactor core isolation cooling (RCIC) and RHR 'A', 'B' and 'C' pump rooms and entry into EOP-3, "Secondary Containment Control." This review included procedures and drawings used, immediate and long term corrective actions, and the cause analysis performed by the licensee. Documents reviewed in this inspection are listed in the Attachment.

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

b. Findings

Introduction: A finding of very low safety significance (Green) and associated NCV of TS 5.4.1 was self-revealed when plant operators failed to follow a written procedure. Specifically, operators did not verify expected plant results while manipulating plant equipment which resulted in draining approximately 15,000 gallons from the suppression pool, causing unexpected alarms for RCIC, and RHR 'A', 'B', and 'C' pump room sumps and water intrusion into the Auxiliary and Intermediate Buildings.

Description: On May 19, 2011, maintenance personnel were performing SVI-E12-T2023, "RHR B&C Waterleg Pump and Check Valve Cold Shutdown Operability Test" in order to establish a pump curve for a newly installed waterleg pump. In order to establish the desired flow rates to support development of a pump curve, operators were

directed by procedure to open valves on two 3/4-inch lines and eventually an 8-inch line which drains to the Auxiliary Building sump. Subsequent to opening the 8-inch line valve, high level alarms were received for the RCIC and RHR 'A,' 'B,' and 'C' ECCS pump room sumps. This prompted the licensee to enter EOP-3, “Secondary Containment Control,” which directed the licensee to secure all sources of water to the sump.

The licensee subsequently determined that opening the 8-inch valve caused a pressure drop in the RHR system which allowed an associated line check valve to open and subsequent draining of the suppression pool to the Auxiliary Building sump. This caused the Auxiliary Building sump to overflow resulting in water intrusion to various areas of the Auxiliary and Intermediate Buildings. Reactor core isolation cooling and RHR 'A', 'B', and 'C' pump room drain isolation valves were open at this time which allowed water from the Auxiliary Building sump to enter the ECCS rooms via the drains. The ECCS room sump alarms were the entry condition which prompted the licensee to enter EOP 3. The licensee estimated that the 8-inch valve was open for approximately 10 minutes and that approximately 15,000 gallons was drained from the suppression pool. This water significantly contaminated various portions of the Auxiliary and Intermediate Buildings.

In the past, the licensee was able to achieve the desired flow rates for the test with only the two 3/4-inch drain lines. Due to a new standard that requires more data points, the 8-inch valve was opened per procedure to obtain higher flow rates than previously required for the test. The step for opening the 8-inch valve had previously been incorporated into SVI-E12-T2023 but had never before been utilized to complete the testing. During the pre-job brief, this step was not expected to be needed and was briefed as a contingency plan. The operator opening the valve heard an unexpected large volume of water flowing through the pipe but continued to open the valve until fully opened because the desired flow rate through the pump had not been achieved. This is contrary to the licensee’s Normal Operating Procedure (NOP)-OP-1002, “Conduct of Operations,” Revision 5, step 4.3.2.2, which states in part, “Anticipate the impact of component operation prior to its operation, and then verify that the expected effects occur during and following the operation.”

Analysis: The inspectors determined that the licensee’s failure to verify expected effects during manipulation of plant components constituted a performance deficiency warranting a significance evaluation in accordance with IMC 0612, Power Reactor IRs, Appendix B, "Issue Screening.” This performance deficiency was not similar to any of the examples in IMC 0612, Appendix E, “Examples of Minor Issues,” but was characterized as more than minor because it was associated with the Human Performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations.

The inspectors performed a significance determination of this issue using IMC 0609, “Significance Determination Process,” IMC 0609.04, “Initial Screening and Characterization of Findings,” and IMC 0609, Appendix G, Attachment 1, “BWR Refueling Operation with RCS Level >23’.” The issue screened as an internal/external flooding initiator contributor. The finding was of very low safety significance because it did not increase the likelihood of a loss of RCS inventory, degrade the licensee’s ability

to terminate a leak path or add RCS inventory, or degrade the licensee's ability to recover DHR.

The finding has a cross-cutting aspect in the Resources component of the Human Performance cross-cutting area per IMC 0310 (H.2(c)) because the licensee failed to provide complete, accurate and up-to-date procedures. Specifically, the procedure to test the HPCS waterleg pump did not adequately address the potential to drain the suppression pool to the Auxiliary Building sump.

Enforcement: Technical Specification 5.4.1 requires that written procedures/instructions be established, implemented, and maintained for applicable procedures described in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, February 1978. The administrative procedures section of RG 1.33 includes procedures for authorities and responsibilities for safe operation and shutdown. Contrary to the above, the licensee did not follow NOP-OP-1002. Specifically, the licensee did not verify that the expected effects occurred when opening a larger flow path to complete a test and continued to open the 8-inch drain line after unexpectedly high volumes of flow were heard in the line. The water flow ultimately resulted in high sump level alarms in the RCIC and RHR 'A,' 'B,' and 'C' pump rooms as well as flooding in the Auxiliary Building ECCS corridor. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CR 11-95107, this violation is being treated as an NCV, consistent with section 2.3.2 of the NRC Enforcement Policy. (NCV 05000440/2011003-03, Failure to Verify Expected Effects Results in Overflowing the Auxiliary Building Sump)

.3 Turbine Control Valve 4 Closure

a. Inspection Scope

The inspectors reviewed the plant's response to a closure of turbine control valve 4 on June 21, 2011. The control valve closure resulted in several steam bypass valves opening to maintain reactor pressure. The licensee reduced reactor power to 75 percent power to close the bypass valves and maintain the plant stable. The inspectors monitored the licensee's response and reviewed the planned corrective actions for this event. The licensee entered the event into their CAP for evaluation as CR 11-96684, repaired the valve, and returned to 100 percent power. Documents reviewed in this inspection are listed in the Attachment.

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

b. Findings

No findings were identified.

4OA5 Other Activities

.1 (Closed) NRC Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event"

The inspectors assessed the activities and actions taken by the licensee to assess its readiness to respond to an event similar to the Fukushima Daiichi nuclear plant fuel damage event. This included (1) an assessment of the licensee's capability to mitigate conditions that may result from beyond design basis events, with a particular emphasis

on strategies related to the spent fuel pool, as required by NRC Security Order Section B.5.b issued February 25, 2002, as committed to in severe accident management guidelines (SAMGs), and as required by 10 CFR 50.54(hh), (2) an assessment of the licensee's capability to mitigate station blackout conditions, as required by 10 CFR 50.63 and station design bases, (3) an assessment of the licensee's capability to mitigate internal and external flooding events, as required by station design bases, and (4) an assessment of the thoroughness of the walkdowns and inspections of important equipment needed to mitigate fire and flood events, which were performed by the licensee to identify any potential loss of function of this equipment during seismic events possible for the site.

Inspection Report 05000440/2011011 (ADAMS Accession Number ML111320382) documented detailed results of this inspection activity. Following issuance of the report, the inspectors conducted detailed follow-up on selected issues.

.2 (Closed) NRC Temporary Instruction 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)"

On May 17, 2011, the inspectors completed a review of the licensee's SAMGs, implemented as a voluntary industry initiative in the 1990's, to determine (1) whether the SAMGs were available and updated, (2) whether the licensee had procedures and processes in place to control and update its SAMGs, (3) the nature and extent of the licensee's training of personnel on the use of SAMGs, and (4) licensee personnel's familiarity with SAMG implementation.

The results of this review were provided to the NRC task force chartered by the Executive Director for Operations to conduct a near-term evaluation of the need for agency actions following the Fukushima Daiichi fuel damage event in Japan. Plant specific results for Perry Nuclear Power Plant were provided as an Enclosure to a memorandum to the Chief, Reactor Inspection Branch, Division of Inspection and Regional Support, dated June 1, 2011 (ADAMS Accession Number ML111520396).

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

a. Inspection Scope and Documentation

As documented in Section 1R04, the inspectors confirmed the acceptability of the described licensee's actions. This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later IR.

As documented in Section 4OA2, the inspectors confirmed the acceptability of the described licensee's actions. This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later IR.

On May, 6, 2011, the inspectors conducted a walkdown of normally inaccessible portion of piping of LPCS in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspectors' independent walkdown (TI 2515/177, Section 04.02.c.3).

In addition, the inspectors verified that the licensee had isometric drawings that describe the LPCS system configurations and had acceptably confirmed the accuracy of the drawings (TI 2515/177, Section 04.02.a). The inspectors verified the following related to the isometric drawings:

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

The inspectors verified that P&IDs accurately described the subject systems, that they were up to date with respect to recent hardware changes, and any discrepancies between as-built configurations, the isometric drawings, and the P&IDs were documented and entered into the CAP for resolution (TI 2515/177, Section 04.02.b).

Documents reviewed are listed in the Attachment to this report.

This inspection effort counts towards the completion of TI 2515/177 which will be closed in a later IR.

b. Findings

No findings were identified.

4OA6 Meetings

.1 Exit Meeting Summary

On July 11, 2011, the inspectors presented the inspection results to the Site Vice President, Mr. Mark Bezilla, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary. Some proprietary material was reviewed by the inspectors during the RFO 13 inspection process. The material was returned to the licensee at this meeting.

.2 Interim Exit Meetings

Interim exits were conducted for

- the results of the inservice inspection with Plant General Manager, Mr. K. Krueger on April 29, 2011; and
- radiological hazard assessment and exposure control program and ALARA planning and controls program with Mr. M. Bezilla, and other licensee staff on May 6, 2011.

The inspectors confirmed that none of the potential report inputs discussed were considered proprietary.

4OA7 Licensee-Identified Violations

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

- Title 10 CFR Part 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" requires the licensee to assess and manage the increase in risk that may result from proposed maintenance activities. Contrary to the above, on May 24, 2011, it was identified that the licensee had an unplanned entry into yellow shutdown safety risk when planned maintenance caused two source range monitors to become unavailable. The issue was documented in the licensee's CAP as CR 11-95282. Corrective actions included inclusion into training as operating experience and detailed review of applicable procedures.

  The failure to assess the increase in risk is a performance deficiency as defined in IMC 0612, "Power Reactor IRs," Appendix B, "Issue Screening." The inspectors determined that the finding was more than minor because it is similar to example 7.e of IMC 0612, Appendix E, "Examples of Minor Issues" and resulted in the qualitative shutdown probabilistic risk assessment (PRA) risk crossing the threshold into a higher licensee-established risk category. Therefore, the performance deficiency is associated with the Mitigating Systems Cornerstone attribute of Equipment Performance and adversely impacted the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was reviewed for significance in accordance with IMC 0609, Appendix K, "Maintenance Risk Assessment and Risk Management Significance Determination Process." Appendix K directs that for licensees who only perform qualitative analyses of plant configuration risk due to maintenance activities, as is the case during shutdown for this licensee, the significance of the deficiencies must be determined by an internal NRC management review using risk insights where possible in accordance with IMC 0612. The NRC management review concluded that this finding was of Green safety significance because missing risk management actions did not result in loss of key shutdown risk functions and plant TS continued to be met for the plant conditions at the time.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Bezilla, Vice President Nuclear
K. Krueger, Plant General Manager
D. Evans, Work and Outage Management Director
J. Grabnar, Site Engineering Director
H. Hanson, Performance Improvement Director
T. Jardine, Operations Manager
P. McNulty, Radiation Protection Manager
M. Stevens, Maintenance Director
J. Tufts, Chemistry Manager

NRC

N. Valos, Senior Reactor Analyst
D. Passehl, Senior Reactor Analyst

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Opened and Closed

05000440/2011003-01	NCV	Failure to Establish a Procedure to Operate Safety-Related Equipment (Section 1R15)
05000440/2011003-02	FIN	Failure to Follow TS Bases (Section 1R15)
05000440/2011003-03	NCV	Failure to Verify Expected Effects Results in Overflowing the Auxiliary Building Sump (Section 4OA3.1)

Discussed

2515/183	TI	Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event ADAMS Accession Number ML111320382
2515/184	TI	Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs) ADAMS Accession Number ML111520396
2515/177	TI	System Walkdown Associated with TI 2515/177, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems

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 IR.

1R01 Adverse Weather

- Perry Answer to Question on Roof Loading for Clogged Drains on Control Complex and Intermediate Building Roofs
- NOP-WM-2001; Work Management Scheduling/Assessment/Seasonal Readiness Processes; Revision 11
- IOI-15; Seasonal Variations; Revision 18
- Listing of Summer Preparation Work Orders January 2011 through May 2011; dated June 19, 2011
- CR 10-79529; P46 Chiller Refrigerant Leaks; dated July 10, 2011
- WO 200439600; Clean, Inspect, Rework Chiller Condenser; dated April 7, 2011
- PAP-0102; Interface with the Transmission System Operator; Revision 7
- Perry System Health Report; R11 – Station Transformers; 2011-1 (1st quarter)
- Perry System Health Report; S42 – Transmission Station; 2011-1 (1st quarter)
- ONI-R10; Loss of AC Power; Revision 9
- ONI-S11; Hi/Low Voltage; Revision 8

1R04 Equipment Alignment

- SOI-E21; Low Pressure Core Spray; Revision 24
- VLI-E21; Valve Lineup Instruction – Low Pressure Core Spray; Revision 8
- Drawing D-304-707; Low Pressure Core Spray Reactor Building Plan; Revision K
- Drawing D-304-707 Isometric Drawing 1-E21-1; Low Pressure Core Spray Reactor Building; Revision J
- Drawing D-304-707 Isometric Drawing 1-E21-2; Low Pressure Core Spray Reactor Building; Revision J
- Perry Drywell Pipe Slope Report for LPCS; dated April 23, 2009
- Drawing 302-701-00000; High Pressure Core Spray System; Revision HH
- VLI-E22A; High Pressure Core Spray; Revision 9
- VLI-C41; Stand Liquid Control System; Revision 8
- Drawing 302-691; Standby Liquid Control System; Revision V
- CR 11-96731; NRC-Identified Valve Closed But Not Locked as Required; dated June 21, 2011
- SVI-T23-1202; Monthly Primary Containment and Drywell Integrity Verification; Revision 3
- VLI-M23/24; MCC, Switchgear and Miscellaneous Electrical Equipment Area HVAC System; Revision 7
- Drawing 912-609; Switchgear and Misc. Electrical Equipment Area HVAC System and Battery Room Exhaust; Revision AA

1R05 Fire Protection (Annual/Quarterly)

- PAP-1910; Fire Protection Program; Revision 23
- PAP-0204; Housekeeping/Cleanliness Control Program; Revision 25
- FPI-0CC; Control Complex; Revision 8

- Drawing 105-0011-00000; Control Complex Floor Plan - El. 574'-10"; Revision J
- FPI-0FH; Fuel Handling Building; Revision 4
- FPI-0IB; Intermediate Building; Revision 6
- Drawing 101-0033-00000; Intermediate Building Floor Plan - El. 620'-6"; Revision K
- FPI-1RB; Reactor Building; Revision 4
- Drawing 101-0051-00000; Reactor Building Floor Plan – El. 574'-10" and 599'-9", Revision F
- Drawing 101-0052-00000; Reactor Building Floor Plan – El. 620'-6" and 642'-0", Revision F

1R06 Flood Protection Measures

- CR 11-92019, Turbine Building to Heater Bay Wall Penetration Blocked; dated March 30, 2011
- CR 11-92037, Potential Flooding Concern HB 620; dated March 30, 2011
- Drawing 911-0617; Auxiliary Building Drains; Revision F

1R07 Heat Sink

- Maintenance Monitoring Report for 1P42B0001B Heat Exchanger; dated April 27, 2011
- WO 200415917; RHR Heat Exchangers A and C Performance Testing; dated November 5, 2010
- GMI-0023; In-Field Reference for Inspection of Fresh Water Systems for Clams and Mussels; Revision 4; dated August 10, 2006
- WO 200342060; Eddy Current, Clean and Inspect ECC B Heat Exchanger; dated April 26, 2011

1R08 Inservice Inspection Activities

- CR 11-90659, Additional 1R13 Steam Dryer Inspection Scope Identified; dated March 7, 2011
- CR 11-88606, Emergent Work/Schedule Delay/Impact; dated January 23, 2011
- CR 11-93275, Contractor (GE) Welder Performed Welding With Expired Qualifications; dated April 22, 2011
- NQI-0964, Procedure For Ultrasonic Examination of Reactor Pressure Vessel Welds; Revision 0
- NQI-1042, Visual Examination; Revision 14
- Weld History Record (Single Joint), Work Order 200262683; dated April 6, 2009
- Weld Rod Issue Receipt 0900956; dated April 6, 2009
- Welding Procedure Specification, 1.1.2-001; Revision 11

1R11 Licensed Operator Requalification Program

- OTLC-JITSU114_PY-SG1; Just-In-Time Training for Post-RFO 13, Plant Startup 114; Revision 1; dated May 25, 2011
- Start Up Reactivity Plan for Startup 114; Revision 0

1R12 Maintenance Effectiveness

- WO 200282210; Replace Piston/Rods/Liners/Heads/Air Start Valves; dated May 1, 2011
- WO 200458482; Replace or Refurbish Pistons; dated May 6, 2011
- Drawing 302-0348-00000; Standby Diesel Engine Mounted Piping 1R43-C0001B Div.2; Revision H
- CR 11-93386; Bolting for Diesel Cylinder Heads, Bolts That Exceeded Break Away Value; dated April 25, 2011

- CR 11-93881; Division 2 Diesel Generator Tin Flakes Found in Crankcase; dated May 1, 2011
- CR 11-94208; Regulator Found to be Mounted Incorrectly; dated May 5, 2011
- CR 11-94266; Division 2 Diesel Generator Tin Flakes Found Post 15 Min Run; dated May 6, 2011
- SVI-P53-T7312; Upper Containment Airlock Pneumatic System Leak Test, Pen#312; Revision 14
- CR 11-96616; Upper Containment Airlock Inner Door Large Seal Pneumatic Decay Test – Leakage; dated June 18, 2011
- WO 200363557; (92D) Penetration Pressurization Valve Operability Test; dated June 11, 2011
- WO 200374030; Lubricate – Upper Containment – Inner Air Lock Door; dated June 8, 2011
- CR 08-44244; Issues Identified While Troubleshooting Airlock; dated August 4, 2008
- WO 200346680; Rebuild Ball Valves/Inspect Mechanical Interlocks; dated June 21, 2011
- WO 200191413; Rebuild Ball Valves/Inspect Mechanical Interlocks; dated June 14, 2008
- GMI-0176; Containment Airlock Door Maintenance; Revision 9

1R13 Maintenance Risk Assessments and Emergent Work Control

- Perry NPP 1R13 Pre-Outage Defense-In-Depth Report; dated March, 18, 2011
- NOP-OP-1005; Shutdown Defense In Depth; Revision 12
- NOP-OP-1007; Risk Management; Revision 8
- PYBP-POS-2-2; Protected Equipment Postings; Revision 10
- Form PNPP-10203; Protected Equipment Posting Checklist; Revision date 9/16/09
- CR 11-94269; Rigging Clamp Failed While Rigging CRD Mech From DW 599 to DW 583; dated May 6, 2011
- TXI-0393; Control Rod Drive Mechanisms (CRDM) Using Motion Assisted Drive Exchange Assembly (MADEA); Revision 0
- CR 11-95282; Actions Initiated to Remove SRM D From Service Ahead of Schedule; dated May 21, 2011
- CR 11-95557; Inappropriate Human Performance Success Clock Evaluation Based on Shutdown Safety Risk Increase; dated May 26, 2011
- Operational Decision Making Issue Evaluation for CR 11-95409; dated May 25, 2011
- CR 11-95409; Documentation of ODMI for I&C SVI Performance During SDC Operations; dated May 24, 2011
- Operations Evolution Order (OEO) to Jumper High Pressure Signal to Prevent Shutdown Cooling Isolation Signal When in Modes 4 or 5; dated May 25, 2011
- Regulatory Applicability Determination No. 11-02267; Residual Heat Removal System Revision 53 and OEO to Jumper Reactor Pressure Vessel Shutdown Cooling Pressure Trip in Modes 4 or 5; dated May 25, 2011

1R15 Operability Evaluations and Functionality Assessments

- CR 11-91273; Potential Non-Conservative Technical Specification Application; dated March 18, 2011
- CR 11-94572; RHR Shutdown Cooling Isolation Valves; dated May 11, 2011
- CR 11-95409; Documentation of ODMI for I&C SVI Performance During SDC Operations; dated May 24, 2011
- Operations Standing Order; Shutdown Cooling Operations; dated April 18, 2011
- NOP-OP-1002; Conduct of Operations; Revision 05
- SOI-E12; Residual Heat Removal System; Revision 55
- CR 11-94382; 1N27F0514A, Body Seat Degraded; dated May 9, 2011
- NOP-OP-1010; Operational Decision-Making; Revision 3

- CR 11-96323; Recirc Pump B Lower Bearing Low Flow Alarm; dated June 12, 2011
- CR 11-96328; Annunciator Rcirc B Outer Seal Leakage Hi is Locked In; June 12, 2011
- Operational Decision Making Issue Evaluation for CR 11-96323 and CR 11-96328; dated June 23, 2011

1R18 Permanent Modifications

- WO 200375169; Replace MCC F1C08 Auto XFR SW
- ECP 10-0266-01, Rev 2; Upgrade the Existing 480-Volt Motor Control Center (MCC) Automatic Transfer Switch and Relay in MCC F1C08
- CR 11-95294; Unable to Transfer F1C08 to Normal Source; dated May 23, 2011
- CR 09-60892; MCCs F1B08, F1C08 and F1D08 Transferred to Their Emergency Source; dated June 13, 2009

1R19 Post-Maintenance Testing

- WO 200416219; Replace SRM 'C' Detector; dated April 20, 2011
- ICI-C-C51-6; Source Range Monitor / Intermediate Range Monitor Detector Drive Mechanism Calibration Data Sheet; Revision 1
- SVI-C51-T0234-A; SRM A and C Channel Functional Test for 1C51-K600A and 1C51-K600C; Revision 7
- WO 200329027; New PM – Replace Degraded Voltage Control Relays in EH13 (Div. 3); dated April 28, 2011
- SVI-E22-T1329; Division 3 HPCS Diesel Generator Functional Test; Revision 11
- SVI-E22-T1339; Division 3 HPCS Diesel Generator Loss of Off-site Power Test; Revision 10
- SVI-E22-T5397; HPCS Initiation and Loss of EH13 Response Time Test; Revision 13
- CR 11-93898; Discrepancies in Local vs. Remote Kilowatt and KVAR Indications, Division 3 DG; dated May 1, 2011
- WO 200282210; Replace Piston/Rods/Liners/Heads/Air Start Valves; dated April 23, 2011
- WO 200330522; DG-Break-In Runs/Hot Webs/Firing Pressures; dated May 5, 2011
- PMI-0032; Division 1&2 Emergency Diesel Generator Crankshaft Web Deflections and Thrust Check; Revision 7
- PMI-0053; Division 1&2 Standby Diesel Generator Connecting Rod and Piston Maintenance; Revision 9
- CMI-0018; Division 1 and 2 Emergency Diesel Generator Cylinder Liner Honing; Revision 4
- CMI-0016; Division 1&2 Emergency Diesel Generator Starting Air Valve Repair; Revision 6
- WO 200458482; Replace Division 2 EDG Pistons and Drain Lube Oil and Jacket Water; dated May 7, 2011
- SVI-E12-T2023; RHR B&C Waterleg Pump and Check Valve Cold Shutdown Operability Test; Revision 9
- WO 200377606; Perform Inspection/Replacement the Rotary Variable Differential Transducer (RVDT) For Recirc Loop ‘B’ Flow Control Valve; dated May 25, 2011
- WO 200463277; Received Unexpected HPU B Shutdown during HPU Startup per SOI-B33; dated June 10, 2011
- WO 200458637; Replace RCIC Minimum Flow Valve; dated May 25, 2011
- WO 200369758; RCIC System functional Pressure Test; dated June 6, 2011
- ISI-E51-T1100-2; Reactor Core Isolation Cooling (RCIC) System Functional Pressure Test – Class 2; Revision 5
- WO200369765; RCIC System low Pressure Operability Test; dated June 6, 2011
- SVI-E51-T1272; RCIC System Low Pressure Operability Test; Revision 15
- SVI-B21-T0246-B; ATWS-RPT Logic System Functional Test for Division 2; Revision 5

- SVI-D17-T8037; Unit 2 Vent Noble Gas Radiation Monitor Calibration for 2D17-K786; Revision 6
- CR 11-96313; Loss of Unit 2 Plant Vent Noble Gas Channel; dated June 10, 2011
- CR 11-91909; Misposition: Inadvertent Shutdown of the Unit 2 Plant Vent Isokinetic; dated March 29, 2011

1R20 Refueling Outage

- NOP-CC-3002-01; Seismic Evaluation of Control Rod Seal in Outage; Revision 3
- Drawing 305-0006-00104; ISI – System B13, RV – Bottom Head Weld Arrangement; Revision C; dated October 18, 2010
- ISI-B21-T1300-1; Reactor Coolant System Leakage Pressure Test; Revision 17; dated March 7, 2011
- 1R13 Updated Defense-in-Depth Report; Revision 2; dated May 10, 2011
- FTI-E0006; Installation and Removal of SRM, IRM, and LPRM Dry Tubes; Revision 5 and 6
- TXI-0395; BWR CRD Mechanism Vertical Maintenance; Revision 0 dated April 28, 2011
- WO 200083188; LPRM Dry Tube Replacement w/Addendum A-1; dated May 15, 2011
- CR 11-94572; RHR Shutdown Cooling Isolation Valves; dated May 11, 2011
- CR 11-91273; Potential Non-Conservative Technical Specification Application; dated March 18, 2011
- CR 11-88749; Operations with the Potential to Drain the Reactor Vessel (OPDRV) Definition; dated January 26, 2011
- EER 600672007; Evaluation for OPDRV Associated with Dry Tube Replacement; dated April 13, 2011
- Global Nuclear Fuels Supplemental Reload Licensing Report for Perry 1, Reload 13 Cycle 14; dated December 2010
- Core Operating Limits Report for the Perry Nuclear Power Plant Unit 1 Cycle 14 (Reload 13); Revision 18
- GMI-0067; Installation and Removal of Control Rod Drives and Thermal Sleeves; Revision 12 dated April 28, 2011
- TXI-0393; Control Rod Drive Mechanisms (CRDM) Using Motion Assisted Drive Exchange Assembly (MADEA); Revision 0 dated April 28, 2011
- CR 11-94006; Foreign Material Found During IVVI Inspections; dated May 3, 2011
- CR 11-94052; Condition Monitoring Exams of Jet Pump 13 & 14 Problems Found in 1R12; dated May 3, 2011
- CR 11-94049; FME Cover Falls Into Feedwater Line; dated May 3, 2011
- CR 11-93343; 1P11-F060 Leakage Found to be Unacceptably High; dated April 23, 2011
- IOI-9; Refueling; Revision 27
- CR 11-94269; Rigging Clamp Failed While Rigging CRD Mechanism From DW 599 to DW 583; dated May 6, 2011
- NOP-WM-5003; Rigging, Lifting and Load Handling; Revision 2
- CR 11-91461; 1R13 Shutdown Defense-In-Depth Snapshot Assessment Area for Improvement; dated March 22, 2011

1R22 Surveillance Testing

- SVI-P42-T2001A; Emergency Closed Cooling System A Pump and Valve Operability Test; Revision 10
- eSOMS Narrative Logs; dated April 4, 2011
- Drawing 302-0621-00000; Emergency Closed Cooling System; Revision SS
- System Description Manual P42; Emergency Closed Cooling (ECC) System; Revision 12

- CR 11-93000; IPTE for Reactor Shutdown Improvement Opportunity; dated April 18, 2011
- WO 200258135; EOC-RPT Breaker Arc Suppression Response Time for 1B33A-CB3A and 1B33A-CB3B; Revision 0; dated April 17, 2011
- SVI-B33-T0257-A; EOC-RPT Breaker Arc Suppression Response Time for 1B33A-CB3A and 1B33A-CB3B; Revision 4
- SVI-F15-T1319; Refueling Platform Operability Test; Revision 12
- SVI-C71-T0427; Rx Mode Switch Refuel Mode Channel Functional; Revision 8
- WO 200366246; Division 2 Standby Diesel Generator Loss of Offsite Power Test; dated May17, 2011.
- SVI-R43-T1338; Division 2 Standby Diesel Generator Loss of Offsite Power (LOOP) Test; Revision 20
- SVI-R43-T5367; LPCI B and C Initiation and Loss of EH12 Response Time Test; Revision 20
- WO 200418519; Lower Primary Containment Air Lock (Penetration P305), in Between the Seals Test; dated June 27, 2011
- SVI-P53-T6305; Lower Primary Containment Air Lock (Penetration P305), in Between the Seals Test; Revision 5
- WO 200418523; Upper Primary Containment Air Lock (Penetration P312), in Between the Seals Test; dated June 27, 2011
- SVI-P53-T6312; Upper Primary Containment Air Lock (Penetration P312), in Between the Seals Test; Revision 6

4OA1 Performance Indicator Verification

- NOBP-LP-4012-08; MSPI Data Sheets for Safety System Functional Failures from April 2010 to March 2011; Revision 2
- NOBP-LP-4012-04; MSPI Data Sheets for Emergency AC Power Systems from April 2010 to March 2011; Revision 2
- NOBP-LP-4012-05; MSPI Data Sheets for High Pressure Injection System from April 2010 to March 2011; Revision 2
- Mitigating Systems Performance Index Basis Document; Revision 4
- NOBP-LP-4012; NRC Performance Indicators; Revision 3
- PYBP-DES-0011; Mitigating Systems Performance Index; Revision 1

4OA2 Identification and Resolution of Problems

- Root Cause Analysis Report: Preparations for Online Work Windows Has Not Met Work Management Expectations or Milestones; dated December 22, 2010
- CR 10-85080; Preparations for the Recent Divisional Outages Has Had Less Than Acceptable Adherence With The Work Management Process Expectations and Compliance With The Schedule Milestones; dated October 28, 2010
- PR 10-85080-7; Cycle Schedule Resource Levelization Action Plan; dated December 20, 2010
- PR 10-85080-8; Maintenance/Training Action Plan; dated December 20, 2010
- CR 11-91031; During the Review of CR 10-85080, Several Issues with the Documented Causes and Corrective Actions Were Identified; dated March 15, 2011
- CR 11-91029; During the Review of CR 10-85080, Several Issues with the Operating Experience Review Were Identified; dated March 15, 2011
- CR 09-62307; PYWM IPA Addresses Work Groups Not Adequately Preparing to Support the Work Implementation Schedule as Published; dated July 24, 2009

- CR 10-83457; The Effectiveness Evaluation Performed on 8/3/10 of CR 09-62307 (titled: PYWM IPA Addresses Work Groups Not Adequately Preparing to Support the Work Implementation Schedule as Published) Identified That Collectively the Corrective Actions Were Ineffective in Resolving the Condition; dated September 30, 2010
- CR 10-85282; PY-PA-10-04 Escalation of Shortfalls in Site Support of Schedule Development; dated November 1, 2010
- CR 10-85563; Effectiveness Review Rated as Ineffective for CR 08-4821 / CA 09-59856-15; dated November 9, 2010
- CR 09-59856; PY-PA-09-01 Finding: Work Management Program Rated as Ineffective (Red); dated May 29, 2009
- CR 10-81038; Implementation of the Work Management Process; dated August 10, 2010
- NOBP-LP-2011; FENOC Cause Analysis; Revision 12
- NOP-LP-2001; Corrective Action Program; Revision 27
- NOBP-LP-2007; Condition Report Process Effectiveness Review; Revision 6
- NOP-WM-0001; Work Management Process; Revision 6
- NOP-WM-2001; Work Management Scheduling/Assessment/Seasonal Readiness Processes; Revision 11
- CR 11-90863; Air Observed During RHR C LPCI SVI; dated March 11, 2011
- CR 11-94417; SVI-E12T1182B Air Observed From Vent 1E12F0604B and 1E12F0400B; dated May 9, 2011
- GL 2008-01 Slope Report; dated December 13, 2008
- NRC Generic Letter 2008-01, 9-Month Response – Evaluation Results, Revision 1; dated October 2008
- CR 08-45175; Tabulation of Generic Letter 2008-01 Corrective Actions; dated August 22, 2008
- CR 09-62389; Air Accumulation Criteria for Sensing Lines Referenced to Atmospheric Pressure; dated July 27, 2009
- CR 09-61683; New Void Acceptance Guidance; dated July 10, 2009
- CR 08-50421; Potential Air Pocket in RHR C Loop; dated December 5, 2008
- CR 08-50448; Potential Air Pockets in LPCS; dated December 5, 2008
- CR 08-50401; Potential Air Pockets in RHR B Loop; dated December 4, 2008
- CR 08-50368; Potential Air Pockets in RHR A Loop; dated December 4, 2008
- CR 08-50049; Potential HPCS Air Pocket; dated November 25, 2008
- SVI-E12-T1182-C; RHR C LPCI Valve Lineup Verification and System Venting; Revision 14

<u>4OA3 Follow-up of Events and Notices of Enforcement Discretion</u>

- CR 11-93252; 5 Workers Receive Dose Rate Alarms; dated 4/21/2011
- RWP 116013; RFO-13 Undervessel Activities; Revision 1
- ALARA Plan 11-6013; RFO-13 Undervessel Activities; Revision 0
- RWP 116037; RFO-13 1C51K0600C SRM Cable Reinsertion/Securing and RP Support – Emergent Work; Revision 0
- ALARA Plan 11-6037; RFO-13 1C51K0600C SRM Cable Reinsertion/Securing and RP Support – Emergent Work; Revision 0
- Drawing 302-0643-00000; Residual Heat Removal System; Revision ZZ
- Drawing 921-0617-00000; Auxiliary Building Floor & Equipment Drains; Revision G
- Drawing 911-0617-00000; Auxiliary Building Drains; Revision F
- SVI-E12-T2023; RHR B&C Waterleg Pump and Check Valve Cold Shutdown Operability Test; Revision 9
- NOP-OP-1002; Conduct of Operations; Revision 05
- CR 11-95107; Aux Building Flooded During RHR Water Leg Pump Test; dated May 19, 2011
- EOP-3; Secondary Containment Control and Radioactive Release Control; Revision 0

- SOI-G61; Liquid Radwaste Sumps; Revision 0
- CR 11-96684; Unplanned Power Change and ONI Entry; dated June 21, 2011
- ONI-C51; Unplanned Change In Reactor Power Or Reactivity; Revision 24

4OA5 Other Activities

- SOI-E21; Low Pressure Core Spray; Revision 24
- VLI-E21; Valve Lineup Instruction – Low Pressure Core Spray; Revision 8
- Drawing D-304-707; Low Pressure Core Spray Reactor Building Plan; Revision K
- Drawing D-304-707 Isometric Drawing 1-E21-1; Low Pressure Core Spray Reactor Building; Revision J
- Drawing D-304-707 Isometric Drawing 1-E21-2; Low Pressure Core Spray Reactor Building; Revision J
- Perry Drywell Pipe Slope Report for LPCS; dated April 23, 2009
- NOBP-LP-4003A; FENOC 10 CFR 50.59 User Guidelines; Revision 7
- PAP-0524; Emergency Operating Procedures; Revision 10
- SAG-1; Primary Containment Flooding; Revision 2
- SAG-1 (Chart); Chart – Primary Containment Flooding; Revision 4
- SAG-2; RPV, Containment, and Radioactivity Release Control; Revision 2
- SAG-2 (Chart); Chart – RPV, Containment, and Radioactivity Release Control; Revision 4

LIST OF ACRONYMS USED

AC	alternating current
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	*Code of Federal Regulations*
CR	condition report
CRDM	control rod drive mechanism
ECC	emergency closed cooling
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOC-RPT	end of cycle, recirculation pump trip
FENOC	FirstEnergy Nuclear Operating Company
FIN	Finding
HPCS	high-pressure core spray
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation & control
IEL	initiating event likelihood
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
ISI	Inservice Inspection
LCO	limiting condition for operation
LERF	large early release frequency
LOCA	loss-of-coolant accident
LOI	loss of inventory
LOOP	loss of offsite power
LPCS	low-pressure core spray
MCC	motor control center
NCV	non-cited violation
NEI	Nuclear Energy Institute
NOP	Nuclear Operating Procedure
NOS	Nuclear Oversight
NRC	Nuclear Regulatory Commission
ONI	Off-Normal Instruction
OSP	outage safety plan
P&IDs	Piping and Instrumentation Diagrams
PI	performance indicator
PM	post-maintenance
POS	plant operational state
PR	Prevent Recurrence
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RFO	refueling outage
RHR	residual heat removal
RPV	reactor pressure vessel
SAMG	Severe Accident Management Guidelines
SBLC	standby liquid control
SDC	shutdown cooling
SDP	Significance Determination Process
SRA	Senior Reactor Analyst

SRM	source range monitor
TI	Temporary Instruction
TSO	Transmission System Operator
TS	Technical Specification
USAR	Updated Safety Analysis Report
UT	Ultrasonic Examination
WO	work order

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Sincerely,

/RA/

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

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Letter to M. Bezilla from J. Cameron dated July 28, 2011

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