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

July 30, 2009

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

SUBJECT: PERRY NUCLEAR POWER PLANT NRC INTEGRATED INSPECTION REPORT 05000440/2009003

Dear Mr. Bezilla:

On June 30, 2009, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Perry Nuclear Power Plant. The enclosed report documents the inspection findings which were discussed on July 15, 2009, with you and members of your staff.

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

Based on the results of this inspection, one NRC-identified finding and eight self-revealed findings of very low safety significance was identified (green). Seven of the nine findings involved violations of NRC requirements. Additionally, three licensee-identified violations are listed in Section 4OA7 of this report. However, because of the very low safety significance and because the issues were entered into your corrective action program, the NRC is treating these issues as non-cited violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of any non-cited violation, or disagree with the characterization of any cross-cutting aspect of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001; with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspectors' Office 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 and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000440/2009003
w/Attachment: Supplemental Information

cc w/encl: J. Hagan, President and Chief Nuclear Officer - FENOC
J. Lash, Senior Vice President of Operations and Chief Operating Officer - FENOC
D. Pace, Senior Vice President, Fleet Engineering - FENOC
K. Fili, Vice President, Fleet Oversight - FENOC
P. Harden, Vice President, Nuclear Support
Director, Fleet Regulatory Affairs - FENOC
Manager, Fleet Licensing - FENOC
Manager, Site Regulatory Compliance - FENOC
D. Jenkins, Attorney, FirstEnergy Corp.
Public Utilities Commission of Ohio
C. O'Claire, State Liaison Officer, Ohio Emergency Management Agency
R. Owen, Ohio Department of Health

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

Sincerely,

/RA/

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

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

Enclosure: Inspection Report 05000440/2009003
w/Attachment: Supplemental Information

cc w/encl: J. Hagan, President and Chief Nuclear Officer - FENOC
J. Lash, Senior Vice President of Operations and
Chief Operating Officer - FENOC
D. Pace, Senior Vice President, Fleet Engineering - FENOC
K. Fili, Vice President, Fleet Oversight - FENOC
P. Harden, Vice President, Nuclear Support
Director, Fleet Regulatory Affairs - FENOC
Manager, Fleet Licensing - FENOC
Manager, Site Regulatory Compliance - FENOC
D. Jenkins, Attorney, FirstEnergy Corp.
Public Utilities Commission of Ohio
C. O'Claire, State Liaison Officer, Ohio Emergency Management Agency
R. Owen, Ohio Department of Health

DOCUMENT NAME: G:\Perry\Perry 2009 003.doc
☐ Publicly Available ☐ Non-Publicly Available ☐ Sensitive ☐ Non-Sensitive
To receive a copy of this document, indicate in the concurrence box "C" = Copy without attach/encl "E" = Copy with attach/encl "N" = No copy

OFFICE	RIII			
NAME	JCameron:dtp			
DATE	07/30/09			

OFFICIAL RECORD COPY

Letter to M. Bezilla from J. Cameron dated July 30, 2009

SUBJECT: PERRY NUCLEAR POWER PLANT NRC INTEGRATED INSPECTION REPORT 05000440/2009003

DISTRIBUTION:
Susan Bagley
RidsNrrPMPerry
RidsNrrDorlLpl3-2
RidsNrrDirsIrib Resource
Cynthia Pederson
Kenneth O’Brien
Jared Heck
Allan Barker
Jeannie Choe
Linda Linn
DRPIII
DRSIII
Patricia Buckley
Tammy Tomczak
ROPreports Resource

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-440
License No:	NPF-58
Report No:	050000440/2009003
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant, Unit 1
Location:	Perry, Ohio
Dates:	April 1, 2009, through June 30, 2009
Inspectors:	M. Franke, Senior Resident Inspector J. Bashore, Acting Senior Resident Inspector M. Wilk, Resident Inspector M. Marshfield, Resident Inspector, Ginna C. Tilton, Senior Reactor Engineer R. Murray, Reactor Engineer M. Phalen, Health Physicist, DRS D. Jones, Reactor Inspector J. Corujo-Sandin, Reactor Engineer
Observers:	R. Leidy, Ohio Department of Health Bureau of Radiation Protection
Approved by:	Jamnes L. Cameron, Chief Branch 6 Division of Reactor Projects

Enclosure

TABLE OF CONTENTS

SUMMARY OF FINDINGS

IR 05000440/2009003; 04/01/2009 – 06/30/2009; Operability Evaluations; Outage Activities; Surveillance Testing; Access Control to Radiologically Significant Areas; Followup of Events and Notices of Enforcement Discretion.

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

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Initiating Event

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed for the failure to implement the requirements of licensee Normal Operating Procedure NOP-OP-1014, "Plant Status Control," Revision 00. Specifically, operations personnel used a mechanical advantage device to increase the closing force on residual heat removal (RHR) valve 1E12F0010 without evaluating the affect on the valve. The increased closing force, i.e. increased torque, sheared the stem for valve 1E12F0010, preventing the valve from opening and preventing the plant from entering shutdown cooling. As part of their immediate corrective actions, licensee personnel repaired the valve stem operator to restore RHR shutdown cooling and entered the issue into their corrective action program.

  The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of equipment performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Specifically, the failure of the RHR shutdown cooling common suction isolation valve caused both trains of shutdown cooling to be unavailable during shutdown operations. Using IMC 0609, Appendix G, "Shutdown Operation Significance Determination Process," Checklist 7, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(c) because the licensee did not ensure adequate supervisory and management oversight of work activities to ensure nuclear safety. Specifically, supervisors were aware of the use of mechanical advantage devices on the RHR shutdown cooling common suction manual isolation valve and did not ensure an appropriate evaluation was conducted. (Section 1R20)

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when technicians performed maintenance on protected equipment without implementing risk management techniques specified in station procedures. This resulted in a loss of shutdown cooling flow to the reactor coolant system. Specifically, the licensee established Normal Operating Procedure NOP-OP-1005, "Shutdown Defense in Depth," Revision 10, as the implementing procedure to manage risk during shutdown conditions. The licensee failed to implement the significant risk management actions prescribed in procedure NOP-OP-1005 for maintenance on protected equipment. This resulted in a blown fuse in the reactor protection system causing a loss of shutdown cooling flow to the reactor coolant system. The licensee replaced the fuse and restored shutdown cooling. This issue was entered into the corrective action program as CR 09-58110.

  The finding was more than minor because it was associated with the equipment performance attribute of the Initiating Events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the finding resulted in a loss of reactor decay heat removal event while the reactor was shutdown. Using IMC 0609, Appendix G, "Shutdown Operation Significance Determination Process," Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a), because the licensee did not appropriately plan the work activity consistent with nuclear safety, incorporating risk insights, job site conditions, or the need for planned contingencies, compensatory actions, and abort criteria. (Section 4OA3.1)

- Green. A finding of very low safety significance was self-revealed on April 28, 2009, for the failure to follow maintenance procedure PTI-N41-P0002, "Generator Switchgear Protective Relay Trip Test," when electricians performed maintenance on an incorrect relay associated with bus L11. Bus L11 was not previously identified as a protected component related to shutdown cooling. Therefore, no risk management actions were implemented by the licensee. The licensee posted bus L11 as a protected train and repaired the 1R22-Q103A and 86B circuitry. The licensee entered the issue into its corrective action program as CR-09-58187.

  The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of equipment performance and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, had the 1R22-Q103A relay circuitry functioned as designed, a loss of decay heat removal event would have occurred. Using IMC 0609, Appendix G, "Shutdown Operation Significance Determination Process," Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(a), because the licensee did not use error prevention techniques commensurate with the risk of the maintenance activity. (Section 4OA3.2)

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance (Green) and associated non-cited violation of Technical Specification (TS) 5.4.1 was self-revealed when technicians failed to implement actions to prevent shorting of energized electrical components during maintenance. Specifically, during surveillance testing activities, a lifted lead shorted to a test lug causing the reactor core isolation cooling (RCIC) Division 2 logic to trip. The technicians suspended their surveillance procedure and operators restored the RCIC system in accordance with licensee procedures. Operators also verified high pressure core spray (HPCS) was operable. The licensee visually inspected the RCIC system and found no apparent damage. The licensee conducted additional training on the use of error prevention tools and entered the issue into the corrective action program as CR 09 59356.

  The finding was determined to be more than minor because the finding was associated with the Mitigating Systems cornerstone attribute of equipment performance and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the short circuit resulted in the RCIC system being inoperable. The finding was determined to have very low safety significance because it did not represent a loss of system safety function, a loss of safety function of a non-TS train designated as risk-significant for greater than 24 hours, an actual loss of safety function of a single train for greater than its TS-allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance per IMC 0305 H.4(a) because the technician failed to use error prevention techniques, such as self-checking, that are commensurate with the risk of the assigned task. Specifically, he did not use 'STAR' (Stop, Think, Act, Review) during an activity that could render the RCIC system inoperable. (Section 1R22)

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was self-revealed for the failure to implement corrective actions to ensure residual heat removal (RHR) check valve 1E12F0050A seated during plant pressurizations. Specifically, the licensee failed to establish and maintain corrective actions for check valve 1E12F0050A inability to seat under low differential pressure conditions, resulting in the over-pressurization of a section of RHR system piping. As part of the licensee's corrective action, the operators depressurized the RHR system below operating pressure and were revising procedures to ensure the check valve 1E12F0050A seats fully during system pressurization. This issue was entered into the licensee corrective action program by CR 09-58808 and CR 09-58995 and an appropriate permanent corrective action was being evaluated.

  The inspectors determined that the finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, removal of the RHR venting evolution from station procedures resulted in an unexpected over-pressurization which could have resulted in system damage. Using IMC 0609, Appendix G, “Shutdown Operations Significant Determination Process,” Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of problem identification and resolution per IMC 0305 P.1(c), because the organization failed to thoroughly evaluate the impact of modifying a corrective action. Specifically, the licensee failed to thoroughly evaluate the consequences of removing the venting section of a procedure that was a corrective action for the check valve’s inability to seat under low differential pressure conditions. (Section 4OA3.3)

Cornerstone: Barrier Integrity

- Green. A finding of very low safety significance and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed when operators failed to maintain reactor coolant system (RCS) temperature greater than the TS minimum allowable temperature of 70 °F because Integrated Operations Instruction (IOI)-9, “Refueling,” was inadequate. The licensee did not properly control an outage activity in that they failed to ensure the water sprayed into the reactor pressure vessel met temperature requirements. As part of their immediate corrective actions, licensee personnel stopped the activity and restored the RCS system above the TS 3.4.11 temperature requirements of 70 °F. The licensee entered this issue into the corrective action program as CR 09-55397.

  This finding was determined to be more than minor because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, failure to maintain RCS temperature greater than the minimum allowed by TS affected the functionality of this barrier. Using IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process,” Checklist 8, the inspectors determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance using Figure 1. This finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a) because the organization failed to appropriately plan work and coordinate work activities consistent with nuclear safety. Specifically, job site conditions, including environmental conditions which may impact plant structures, systems, and components; were not considered to ensure water sprayed into the RCS would maintain temperature above 70 °F. (Section 1R15.1)

- Green. A finding of very low safety significance was self-revealed when contract personnel failed to follow the FENOC Industrial Safety Manual to control vehicle movement inside the fuel handling building (FHB). Specifically, a Sea-land truck backed into the FHB roll-up door, dislodging the door in the open position. The licensee suspended fuel movements and implemented compensatory measures for containment integrity. The licensee repaired the roll-up door and conducted training with contract and oversight personnel, and entered the issue into the corrective action program as CR 09 56062.

The finding was determined to be more than minor because the finding was associated with Barrier Integrity cornerstone attribute of SSC and barrier performance and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the finding resulted in an event that challenged FHB integrity, which is a functional barrier to fission product release. Inspection Manual Chapter 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria," was used since IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," does not address the potential risk significance of FHB operations. Regional management determined that the finding was of very low safety significance because there was no fuel handling accident during this period. This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(c) because the licensee failed to ensure adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety is supported. (Section 1R15.2)

Cornerstone: Occupational Radiation Safety

- Green. The inspectors identified a finding of very low safety significance and an associated NCV of TS 5.4.1.a for the failure to establish, implement, and maintain adequate written procedures regarding the radiation safety program. The licensee failed to implement procedurally required compensatory measures associated with moving irradiated fuel assemblies. Specifically, workers on the 360 platform were within close proximity of the refueling mast while fuel moves were in progress. As corrective actions, the licensee posted information signs to control access to specific areas of the 360 platform and planned to incorporate more rigorous radiological controls into the governing procedure. The licensee entered the issue into its corrective action program as CR 09-54697.

  The finding was more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of protecting worker health and safety from exposure to radiation, in that, not implementing adequate radiological control may potentially result in unplanned exposures to radioactive material. The finding was determined to be of very low safety significance because it was not an as-low-as-is-reasonably-achievable planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised. The finding was determined to have a cross-cutting aspect in the decision making component of the human performance area in accordance with IMC 0305 H.1(b), because the licensee did not adequately use conservative assumptions in decision making. (Section 2OS1.4)

- Green. The inspectors reviewed a self-revealing finding of very low safety significance and an associated NCV of TS 5.4.7.a for the failure to barricade and conspicuously post a high radiation area on the 599' elevation of the auxiliary building. As corrective actions, the licensee barricaded and conspicuously posted the affected area as a high radiation area and performed gravity flushes of piping with clean water to reduce the ambient dose rates. The licensee entered the issue into its corrective action program as CR 09-55453.

The finding was more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of protecting worker health and safety from exposure to radiation, in that, not barricading and conspicuously posting high radiation areas may result in unnecessary and unplanned radiation exposures to workers. The finding was determined to be of very low safety significance because it was not an as-low-as-is-reasonably-achievable planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised. The finding was determined to have a cross-cutting aspect in the work control component of the human performance area in accordance with IMC 0305 H.3(b), because the licensee did not adequately coordinate work activities. (Section 2OS1.5)

B. <u>Licensee-Identified Violations</u>

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

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period after core fuel reload activities had started and the plant was in Mode 5. The plant entered Mode 2 on May 11, 2009, at 5:19 p.m. and Mode 1 at 3:16 a.m. on May 12. When the unit synchronized to the grid at 1:02 p.m. on May 12, it concluded the 79-day long refueling outage. The plant was at approximately 35 percent power on May 14 when the 'A' reactor recirculation pump tripped at 10:52 p.m. The operators decreased power to 15 percent power on May 16 to recover the 'A' reactor recirculation pump at 11:37 p.m. The plant achieved 100 percent power on May 24 at 1:26 p.m. On June 12, operators reduced power to approximately 20 percent and disconnected the plant from the grid on June 13 to repair a main transformer bushing. The unit reconnected to the grid on June 13 and power ascension began on June 14. The plant achieved 100 percent power on June 17. On June 21 the plant experienced a turbine trip and reactor scram. Plant startup commenced on June 24 and achieved 100 percent power on June 27. The plant remained at 100 percent power for the remainder of the inspection period.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 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:

- the coordination between the TSO and the plant during off-normal or emergency events;
- the explanations for the events;
- the 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:

- the 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;
- the 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;
- a 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
- the 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 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.

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

b. Findings

No findings of significance 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 Updated Final Safety Analysis Report (UFSAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. Specific documents reviewed during this inspection are listed in the Attachment. 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. The inspectors' reviews focused specifically on the turbine building closed cooling system.

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

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04Q)

a. Inspection Scope

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

- low pressure core spray (LPCS) after alternate decay heat removal modification during the week of May 18, 2009;
- Division 2 Emergency Diesel Generator (EDG) after maintenance run during the week of May 18, 2009; and
- control room ventilation M25 supply fan during the week of June 1, 2009.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety Cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system, and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, UFSAR, 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 activities constituted three partial system walkdown samples as defined in IP 71111.04-05.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (Annual/Quarterly) (71111.05AQ)

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:

- Containment;
- Control Room Area;
- Drywell;
- Control Complex Level 3;
- Intermediate Building Level 2; and
- Heater Bay.

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

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

b. Findings

No findings of significance were identified.

1R06 Flooding Protection Measures (71111.06)

a. Inspection Scope

The inspectors reviewed 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 UFSAR, 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 safety-related equipment areas including alternate decay heat removal, LPCS, and emergency service water (ESW) systems 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

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

b. Findings

No findings of significance were identified.

1R07 Heat Sink Performance (71111.07T)

Triennial Review of Heat Sink Performance

a. Inspection Scope

The inspectors reviewed operability determinations, completed surveillances, vendor manual information, associated calculations, performance test results and cooler inspection results associated with the Emergency Closed Cooling (ECC) Train 'A' heat exchanger and the Ultimate Heat Sink (UHS). These heat exchangers were chosen based on their risk significance in the licensee's probabilistic safety analysis, their important safety-related mitigating system support functions and their operating/inspection history.

For the ECC Train 'A' heat exchanger, the inspectors verified that testing, inspection, maintenance, and monitoring of biotic fouling and macrofouling programs were adequate to ensure proper heat transfer. This was accomplished by verifying the test method used was consistent with accepted industry practices, or equivalent, the test conditions were consistent with the selected methodology, the test acceptance criteria were consistent with the design basis values, and results of heat exchanger performance testing. The inspectors also verified that the test results appropriately considered differences between testing conditions and design conditions, the frequency of testing based on trending of test results was sufficient to detect degradation prior to loss of heat removal capabilities below design basis values, test results considered test instrument inaccuracies and differences and verified the tube and shell side heat loads where equal.

In addition, the inspectors verified the condition and operation of the ECC Train 'A' heat exchanger were consistent with design assumptions in heat transfer calculations and as described in the final safety analysis report. This included verification that the number of plugged tubes was within pre-established limits based on capacity and heat transfer assumptions. The inspectors verified the licensee evaluated the potential for water hammer and established adequate controls and operational limits to prevent heat exchanger degradation due to excessive flow induced vibration during operation. In addition, eddy current test reports and visual inspection records were reviewed to determine the structural integrity of the heat exchanger.

The inspectors verified the performance of UHSs and their subcomponents such as piping, intake screens, pumps, valves, etc., by tests or other equivalent methods to ensure availability and accessibility to the in-plant cooling water systems.

The inspectors reviewed the licensee's operation of service water system and UHS. This included the review of the licensee's procedures for a loss of the service water system or UHS and the verification that instrumentation, which is relied upon for decision making, was available and functional. In addition, the inspectors verified that macrofouling was adequately monitored, trended, and controlled by the licensee to prevent clogging. The inspectors verified that the licensee's biocide treatments for biotic control were adequately conducted and the results monitored, trended, and evaluated. The inspectors also reviewed design changes to the service water system and the UHS.

The inspectors reviewed the licensee's performance testing of service water system and UHS results. This included the review of the licensee's performance test results for key components and service water pump and valve operability test results. In addition, the inspectors compared these results to system configuration and flow assumptions during design basis accident conditions. The inspectors also verified that the licensee ensured adequate isolation during design basis events, consistency between testing methodologies and design basis leakage rate assumptions, and proper performance of risk significant non-safety-related functions.

In addition, the inspectors reviewed CRs related to the heat exchangers/coolers and heat sink performance issues to verify that the licensee had an appropriate threshold for identifying issues and to evaluate the effectiveness of the corrective actions. The documents that were reviewed are included in the Attachment to this report.

These inspection activities constituted two heat sink inspection samples as defined in IP 71111.07-05.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11Q)

a. Inspection Scope

On May 29, 2009, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification examinations to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

The inspectors evaluated degraded performance issues in the turbine lube oil system.

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

- implementing appropriate work practices;
- identifying and addressing common cause failures;
- scoping of systems in accordance with 10 CFR 50.65(b) of the maintenance rule;
- characterizing system reliability issues for performance;
- charging unavailability for performance;
- trending key parameters for condition monitoring;
- ensuring 10 CFR 50.65(a)(1) or (a)(2) classification or re-classification; and
- verifying appropriate performance criteria for SSCs/functions classified as (a)(2) or appropriate and adequate goals and corrective actions for systems classified as (a)(1).

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

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

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

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

- repair of valve 1E12F0010 and operations with potential to drain reactor vessel during the month of April 2009;
- reactor startup in Yellow risk during the week of May 11, 2009; and
- diesel fire pump room foam intrusion and diesel unavailability during the week of June 1, 2009.

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

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

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- ESW 'B' repair during the week of April 6, 2009;
- emergency core cooling system seismic water-leg during the week of April 6, 2009;
- FHB during the week of April 1, 2009;
- operation of the reactor pressure vessel (RPV)/RCS below temperature TS requirements during the week of May 11, 2009;
- ESW system American Society of Mechanical Engineers (ASME) code qualification during the week of May 18, 2009; and
- Division 1 EDG, oil ejected from test pet cock of right bank cylinder Number 5 during the week of June 20, 2009.

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

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

b. Findings

(1) RCS Temperature Below Minimum Allowed by Technical Specifications Due to Inadequate Station Procedures

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed when operators failed to maintain RCS temperature greater than the TS minimum allowable temperature because Integrated Operations Instruction (IOI)-9, “Refueling,” was inadequate.

Description: On March 14, 2009, while draining the upper fuel pools inside containment, licensee personnel were spraying the fuel pool walls with water to reduce radiological dose. The reactor was defueled with the RPV head removed. The upper fuel pools communicate with the RCS during refueling operations. The spray system used water from the condensate storage tank (CST) processed through two mixed bed water systems as a water supply. These systems were not preconditioned to maintain spray temperature greater than 70°F. At 9:02 a.m. the RCS temperature, measured by the bottom head drain, dropped below 70°F. Technical Specification 3.4.11 requires the RCS temperature to be maintained greater than 70°F at all times.

The licensee took immediate action, as required by TS 3.4.11 C.1, to restore RCS temperature above 70°F. The RCS temperature was restored on March 15, 2009, at 4:48 a.m. As required by TS 3.4.11 C.2, the licensee conducted an engineering analysis to determine whether the RCS was acceptable for operation. The analysis determined that the RCS was not in a non-conforming condition and was acceptable for future operation.

The licensee's investigation determined there was no clear station procedural guidance for maintaining temperature above 70°F. The licensee's IOI-9, "Refueling," Revision 23, Section 2.12, states that under no circumstances shall RPV temperature be allowed to go below 50°F due to RT_{NDT} limits. In contrast, the minimum RCS temperature allowed by TS 3.4.11, figure 3.4.11-1 is 70°F. The licensee determined that IOI-9 did not provide guidance to operators in maintaining RPV temperature greater than 70°F.

Analysis: The inspectors determined that IOI -9, “Refueling,” was inadequate in that it specified quantitative acceptance criteria for RCS temperature less than 70°F, contrary to requirements of TS 3.4.11, and was a performance deficiency. The inspectors determined that the finding was more than minor because it was associated with the procedure quality attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, failure to maintain RCS temperature greater than the minimum allowed by TS affected the functionality of this barrier.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, “Shutdown Operations Significance Determination Process.” The inspectors used Checklist 8 in Attachment 1 and determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance (Green) using Figure 1.

This finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a) because the organization failed to appropriately plan work and coordinate work activities consistent with nuclear safety. Specifically, job site conditions, including environmental conditions which may impact plant SSCs, were not considered to ensure the RCS temperature would be maintained above 70 °F.

Enforcement: Criterion V of 10 CFR Part 50, Appendix B, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Contrary to the above, IOI-9, "Refueling," Revision 23, was inappropriate to the circumstance in that it specified an allowable minimum RCS temperature of 50°F which would not ensure compliance with the TS 3.4.11 limit of 70°F. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CR 09-55397, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2009003-01)

(2) Failure to Follow Industrial Safety Manual Results in Damage to Fuel Handling Building Roll-up Door

Introduction: A finding of very low safety significance (Green) was self-revealed when contract personnel failed to follow the FENOC Industrial Safety Manual to control vehicle movement inside the fuel handling building (FHB). Specifically, a Sea-land truck backed into the FHB roll-up door, dislodging the door in the open position.

Description: On March 25, 2009, while fuel inspections were in process, a Sea-land truck was exiting the FHB and was under the control of a spotter. The driver took direction from another individual that the vehicle was clear to exit the FHB. The driver proceeded to move the vehicle in reverse to exit the FHB. At this point, the FHB roll-up door was approximately 6 feet from full open. The vehicle impacted the door, causing the door to come off the track on one side. The FHB roll-up door could no longer be operated. The licensee suspended fuel handling inspections until repairs could be completed.

The reactor vessel was completely defueled with all irradiated fuel having been placed in the FHB pool (lower pool) with a time to boil of 17.5 hours. Fuel moves occurring inside the FHB, were conducted in accordance with Standard Operating Instruction (SOI)-F11, "Fuel Handling Platform," Revision 10. Precaution and Limitation 2.1 of this procedure states, "When performing load movements over irradiated fuel in the FHB, ensure one of the following conditions are met: 1) FHB Integrity is set; 2) Containment/FHB Closure Plan is established in accordance with NOP-OP-1005, Shutdown Safety."

The licensee conducted FHB operations using the closure plan specified by NOP-OP-1005, Revision 10, Section 3.2. This section specifies the action necessary to secure a functional barrier to fission product release, and limit the potential for an unmonitored radiological release. This closure plan specified the use of the FHB roll-up door for containment closure and specified compensatory measures if the door became dysfunctional. Compensatory measures include securing fuel movements and staging tarps near the roll-up door. The licensee secured FHB operations and pursued repair of the roll-up door. The door was fully repaired approximately 10 hours later and FHB operations recommenced.

The licensee's investigation, determined that the licensee's security officer was the vehicle's spotter. The vehicle driver assumed that another licensee employee was the spotter. This other employee provided the driver with an okay to move the vehicle without the security officer's knowledge. The investigation concluded that NPS contractors required clear expectations and training in observing vehicle safety procedures and to specify the licensee employee providing oversight of their activities.

The FENOC Industrial Safety Manual, Revision 7, under the section for Vehicle Safety states, "When backing a truck, be assisted by a member of the crew who is in a position to observe the truck's clearances and communicate with the driver." Contrary to this standard, the licensee did not ensure the Sea-land driver operated under the control of a safety spotter.

Analysis: The inspectors determined that the failure to control vehicle movement in the FHB was contrary to the FENOC Industrial Safety Manual and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with Barrier Integrity cornerstone attribute of SSC and barrier performance and affected the cornerstone objective of providing reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, the finding resulted in an event that challenged FHB integrity, which is a functional barrier to fission product release.

Appendix M, "Significance Determination Process Using Qualitative Criteria," of IMC 0609, was used since IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," does not address the potential risk-significance of FHB operations. Regional management determined that the finding was of very low safety significance (Green) because there was no fuel handling accident during this period.

This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(c) because the licensee failed to ensure adequate supervisory and management oversight of work activities, including contractors, such that nuclear safety was supported.

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. Because the finding does not involve a violation of regulatory requirements, is of very low safety significance, and was addressed in the CAP as CR 09-56062, it is identified as finding (FIN) 05000440/2009003-02.

1R19 Post-Maintenance Testing (71111.19)

a. Inspection Scope

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

- Division 1 EDG following outage maintenance;
- RHR common line suction valve repair;
- RPS master trip unit replacement scram discharge volume; and
- Control room ventilation 'B' supply fan.

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

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

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

a. Inspection Scope

The inspectors reviewed the Outage Safety Plan (OSP) and contingency plans for the refueling outage (RFO), which was conducted February 23 – May 12, 2009, 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 decay heat removal 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;
- refueling activities, including fuel handling and sipping to detect fuel assembly leakage;
- 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; and
- 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

Inability to Operate the RHR Common Suction Line Valve

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” was self-revealed for the failure to implement the requirements of Normal Operating Procedure (NOP)-OP-1014, "Plant Status Control," Revision 00. Specifically, operations personnel used a mechanical advantage device to increase the closing torque on valve 1E12F0010 without evaluating the affect on the valve.

Description: On April 8, 2009, Perry was in Mode 5 for refueling operations. Operators were verifying that both trains of RHR shutdown cooling were available to commence reactor cavity drain down. Indications in the control room for the RHR shutdown cooling common suction manual isolation valve, 1E12F0010, indicated an intermediate position. Operators in the field reported the valve was in the full open position. Control room operators dispatched the Fix-It-Now Team to investigate.

On April 9, 2009, the Fix-It-Now Team discovered that valve 1E12F0010 had 18 inches of stem travel with the stem rotating 360 degrees. The valve vendor manual classified this as indication that the valve stem was not connected to the valve disc. It was also determined by the licensee that the valve disc was in the shut position, which would not allow RHR shutdown cooling. The licensee disassembled the valve and confirmed that the valve stem had failed. The licensee assessment determined that the failure was due to a rapid tensile impact overload and that there were no indications of a manufacturing defect. The licensee repaired the valve and restored RHR shutdown cooling availability. The licensee also conducted ultrasonic testing on similar valves that were manipulated in the same manner; no defects were found.

The licensee's analysis determined that the shutdown cooling suction manual isolation valve had been routinely closed tightly with valve wrenches or 'hooks' during each refueling outage since plant startup to meet the acceptance criteria for surveillance testing of containment isolation valves. The analysis concluded that there was no implementing procedure guidance or limitations on using valve tools for mechanical advantage on manually operated valves. Program level procedure NOP-OP-1014, “Plant Status Control,” Revision 00, specified general guidance for using mechanical advantage devices. Implementing procedure OAI-0201, "Operations General Instructions and Operating Practices," Revision 16, translated the guidance to motor operated valves but not specifically to manual valves.

Licensee procedure, NOP-OP-1014, "Plant Status Control," Revision 00, Section 4.14.13 states, "Valve wrenches are used to apply additional mechanical leverage to a valve that is difficult to operate. This additional leverage has the potential to damage a valve or actuator; therefore, valve wrench use should be avoided if possible and evaluated if use is necessary." Contrary to this requirement, the licensee routinely applied mechanical advantage to valve 1E12F0010 but did not evaluate the effect on the valve. As part of the licensee's corrective action, procedures have been revised to specify a limit for the torque that can be placed on the handwheels of manually operated valves.

Analysis: The inspectors determined that the licensee’s failure to implement controls regarding use of mechanical advantage devices on manually operated safety-related valves was contrary to the requirements of NOP-OP-1014 and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of equipment performance and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown operations. Specifically, the inability to operate the RHR common suction isolation valve removed the availablity of two shutdown cooling systems during shutdown operations.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, “Shutdown Operations SDP.” The inspectors used Checklist 7 contained in Attachment 1 and determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance (Green) using Figure 1.

This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(c) because the licensee did not ensure adequate supervisory and management oversight of work activities. Specifically, supervisors were aware of the use of mechanical advantage devices on the RHR common suction manual isolation valve and did not ensure an appropriate evaluation was conducted.

Enforcement: Criterion V of 10 CFR Part 50, Appendix B, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions or procedures and shall be accomplished in accordance with these instructions or procedures.

Contrary to the above, since plant start-up, the licensee has failed to implement requirements of procedure NOP-OP-1014, "Plant Status Control," Revision 00, by not evaluating the use of valve wrenches on manual valves. Because this violation was of very low safety significance and was entered into the licensee's CAP as CR 09-56938, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2009003-03)

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:

- Div 3 loss of offsite power (LOOP)/loss-of-coolant accident (LOCA) routine testing the week of April 6, 2009;
- RCS leak testing following RFO during the week of May 4, 2009;
- routine Division 1 and 2 LOOP/LOCA testing the week of April 20, 2009;
- routine surveillance testing of RCIC the week of May 18, 2009; and
- RCIC Pump and Valve Operability in-service testing May 12 -13, 2009.

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

- did preconditioning occur;
- were the effects of the testing adequately addressed by control room personnel or engineers prior to the commencement of the testing;
- were acceptance criteria clearly stated, demonstrated operational readiness, and consistent with the system design basis;
- plant equipment calibration was correct, accurate, and properly documented;
- as-left setpoints were within required ranges; and the calibration frequency were in accordance with TSs, the Updated Safety Analysis Report (USAR), procedures, and applicable commitments;
- measuring and test equipment calibration was current;
- test equipment was used within the required range and accuracy; applicable prerequisites described in the test procedures were satisfied;
- test frequencies met TS requirements to demonstrate operability and reliability; tests were performed in accordance with the test procedures and other applicable procedures; jumpers and lifted leads were controlled and restored where used;
- test data and results were accurate, complete, within limits, and valid;
- test equipment was removed after testing;

- where applicable for in-service testing activities, testing was performed in accordance with the applicable version of Section XI, ASME code, and reference values were consistent with the system design basis;
- where applicable, test results not meeting acceptance criteria were addressed with an adequate operability evaluation or the system or component was declared inoperable;
- where applicable for safety-related instrument control surveillance tests, reference setting data were accurately incorporated in the test procedure;
- where applicable, actual conditions encountering high resistance electrical contacts were such that the intended safety function could still be accomplished;
- prior procedure changes had not provided an opportunity to identify problems encountered during the performance of the surveillance or calibration test;
- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

This inspection constituted three routine surveillance testing samples, one in-service testing samples, and one reactor coolant system leak detection inspection samples as defined in IP 71111.22, Sections -02 and -05.

b. Findings

Failure to Prevent Contact of Energized Components Renders RCIC System Inoperable

Introduction: A finding of very low safety significance (Green) and associated NCV of TS 5.4.1 was self-revealed when technicians failed to implement actions to prevent contact of energized electrical components during maintenance. Specifically, the reactor core isolation cooling (RCIC) division 2 logic tripped while attempting to lift leads and a test lug.

Description: On May 18, 2009, an instrumentation and controls (I&C) technician was performing surveillance SVI-E31- T5395B, “Reactor Core Isolation Cooling (RCIC) Steam Line High Channel Functional For 1E31-N0684-B,” Revision 2. During performance of this surveillance a test lug on relay 1E51A-K85 made contact with an adjacent terminal test lug. The short circuit tripped the division 2 RCIC isolation logic, causing the RCIC steam supply inboard isolation valve to close. Actuating the isolation logic also caused the RCIC steam supply trip throttle valve to close. This sequence resulted in the RCIC system becoming unavailable and the licensee entered TS 3.5.3 condition ‘A” for RCIC inoperable. The reactor was in mode 1 at approximately 50 percent power at the time of the trip. The technicians suspended their surveillance procedure and operators restored the RCIC system in accordance with licensee procedures. Operators also verified HPCS was operable. The licensee visually inspected the RCIC system and found no apparent damage.

The technician was performing the surveillance test using surveillance instruction SVI-E31-T5395B, “Reactor Core Isolation Cooling (RCIC) Steam Line High Channel Functional For 1E31-N0684-B,” Revision 2. Step 5.1.14 of the procedure required the technician to lift the wire lead associated with the RCIC system isolation function, which was not identified as a critical step. Maintenance procedure NOP-WM-4006, “Conduct of Maintenance,” Revision 2, Section 4.11.2 requires that leads or jumpers that could affect critical system functions or that could have a significant plant impact should be protected from inadvertent contact. During performance of the surveillance, a test lug on relay 1E51A-K85 made contact with an adjacent terminal test lug, causing actuation of the Division 2 RCIC isolation logic. This caused the inboard steam supply valve to close and the RCIC turbine trip-throttle valve to trip.

The licensee's investigation determined that the cause of this event was the technician not adequately using human performance tools while conducting this activity. This maintenance activity involves routine tasks for loosening lugs and lifting leads. General risks associated with lifting leads were pre-briefed prior to performing the task. However, details of the actual work location and risks specific to this surveillance were not discussed. Walkdowns are not performed for surveillance tests and therefore the specific risks associated with this surveillance were not discovered until after the pre-job brief had been performed. The leads to be lifted are located in an area with limited working space and in close proximity to other energized components. Experience and over confidence of the technician led to not using a robust barrier to mitigate the risk of close proximity of test lugs and other energized electrical components. Failure to employ error prevention techniques resulted in the technician loosening the lug on relay 1E51A-K85 too far and subsequently lead to the unexpected contact between electrical components. Corrective actions by the licensee were to conduct additional training to reinforce the use of human performance tools.

Analysis: The inspectors determined that the technicians failed to follow maintenance procedure NOP-WM-4006, “Conduct of Maintenance,” Revision 2, and was a performance deficiency. The inspectors determined the finding could be evaluated using the SDP in accordance with IMC 0609, “Significance Determination Process,” Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of findings,” Table 4a. The inspectors determined that the finding was more than minor because it was associated with the equipment performance attribute of the mitigating systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the technician's actions resulted in the inoperability and unavailability of the RCIC system.

The finding is determined to have very low safety significance (Green) because it did not represent a loss of system safety function, a loss of safety function of a non-TS train designated as risk significant for greater than 24 hours, an actual loss of safety function of a single train for greater than its technical specification allowed outage time, or screen as potentially risk-significant due to a seismic, flooding, or severe weather initiating event. This finding has a cross-cutting aspect in the area of human performance per IMC 0306 H.4(a) because the technician failed to use error prevention techniques, such as effective pre-job briefs and self-checking, commensurate with the risk of the assigned task. Specifically, the pre-job brief did not identify the risk associated with the actual work location and the technician did not use robust barriers to mitigate those risks.

Enforcement: Perry TS 5.4.1 requires that written procedures/instructions shall be established, implemented, and maintained covering the following activities including the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978. Appendix A, part 9a, required the implementation of written procedures for performing maintenance that can affect the performance of safety related equipment. Perry procedure NOP-WM-4006, "Conduct of Maintenance," Revision 2, implementing Regulatory Guide 1.33, Appendix A, part 9, required that leads or jumpers that could affect critical systems be protected from inadvertent contact. Contrary to the above, while performing surveillance SVI-E31- T5395B, "Reactor Core Isolation Cooling (RCIC) Steam Line High Channel Functional For 1E31-N0684-B," Revision 2, the I&C technician failed to implement actions to ensure that the test lugs, relay leads, and terminal bar were protected from inadvertent contact. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CR 09-59356, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2009003-04)

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2OS1 Access Control to Radiologically-Significant Areas (71121.01)

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

a. Inspection Scope

The inspectors reviewed the licensee's Occupational Exposure Control Cornerstone performance indicator (PI) to determine whether the conditions resulting in any PI occurrences had been evaluated and whether identified problems had been entered into the licensee's CAP for resolution.

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

b. Findings

No findings of significance were identified.

.2 Plant Walkdowns and Radiation Work Permit Reviews

a. Inspection Scope

The inspectors assessed the adequacy of the licensee's internal dose assessment process for internal exposures in excess of 50 millirem committed effective dose equivalent. There were no internal exposures greater than 50 millirem committed effective dose equivalent

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

The inspectors walked down and surveyed (using an NRC survey meter) selected work sites and various areas of the plant, including the main turbine deck, the auxiliary building, containment, and the drywell, to verify that the prescribed radiation work permits, procedures, and engineering controls were in place; that licensee surveys, radiological postings and radiological boundary controls were complete and accurate; and that air samplers were properly located.

This inspection constitutes a supplemental sample previously documented in Inspection Report 05000440/2009002.

b. Findings

No findings of significance were identified.

.3 Problem Identification and Resolution

a. Inspection Scope

The inspectors evaluated the licensee's process for problem identification, characterization, and prioritization and verified that problems were entered into the CAP and resolved. For repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution, the inspectors verified that the licensee's self-assessment activities were capable of identifying and addressing these deficiencies.

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

The inspectors reviewed licensee documentation packages for all PI events occurring since the last inspection to determine if any of these PI events involved dose rates in excess of 25 R/hr at 30 centimeters or in excess of 500 R/hr at 1 meter. Barriers were evaluated for failure and to determine if there were any barriers left to prevent personnel access. Unintended exposures exceeding 100 millirem total effective dose equivalent (or 5 rem shallow dose equivalent or 1.5 rem lens dose equivalent) were evaluated to determine if there were any regulatory overexposures or if there was a substantial potential for an overexposure.

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

b. Findings

No findings of significance were identified.

.4 Job-In-Progress Reviews

a. Inspection Scope

The inspectors reviewed radiological work in high radiation work areas having significant dose rate gradients to evaluate whether the licensee adequately monitored exposure to personnel and to assess the adequacy of licensee controls. These work areas involved areas where the dose rate gradients were severe; thereby increasing the necessity of providing multiple dosimeters or enhanced job controls. Additionally, the inspectors independently observed in-field radiological work practices of craft personnel and radiation protection staff associated with, but not limited to suppression pool diving, refueling activities, work on the alternate decay heat removal modification, and work on the main turbine deck.

This inspection constitutes one sample as defined in Inspection Procedure 71121.01-5.

b. Findings

Introduction: A Green NRC-identified finding of very low safety significance and associated NCV of TS 5.4.1.a for the failure to establish, implement, and maintain adequate written procedures regarding the radiation safety program. Specifically, the licensee failed to implement procedures to ensure that an adequate amount of water shielding for workers on the 360 platform was maintained.

Description: The licensee implemented a design change, Engineering Change Package (ECP) No. 05-0186 to the plant prior to the 2007 refueling outage. This change allowed the installation of the 360 platform, a platform that allows in-vessel inspections to occur concurrently with the offload of fuel from the reactor. Prior to the design change, in-vessel inspections occurred after fuel was offloaded from the reactor. The 360 platform consists of watertight tubs (or troughs) comprised of two stationary tubs on the east and west sides of the reactor and one additional tub that travels nearly the entire circumference of the reactor. The platform allows workers to stand 20-22 inches below the water surface of the reactor cavity, which provides enough head room for workers to continue activities while the refueling bridge passed over the heads and tools of workers performing in-vessel inspections. However, the platform reduced the amount of water shielding between workers in the tubs and the irradiated fuel bundles in the reactor cavity.

Operations Requirements Manual (ORM) Section 6.5.4.1.c established a minimum of 85 inches of water shielding between any worker and the irradiated reactor fuel. The ECP identified this condition and established an administrative horizontal limit of 5 feet between the worker and refueling mast. By maintaining the compensatory limit of 5 feet on the horizontal plane, the licensee maintained the required 85 inches of water between the top of the irradiated fuel bundle and the workers in the tubs. The licensee identified the horizontal 5-feet administrative control as a precaution and limitation (Step 2.33) in procedure SOI-F15 “Refueling and 360 Platforms.” Additionally, a work boundaries map was provided in Attachment 5 of this procedure.

During the inspection, the inspectors identified workers in the 360 platform tubs were within 5 feet of the mast while irradiated fuel was on the grapple at the bottom of the mast. The inspectors were told that the workers were using cameras to assist inspection of the fuel, as the camera(s) of the refueling bridge were not functioning. However, the licensee could not identify to the inspectors any additional administrative controls necessary to ensure that adequate water shielding was maintained. Licensee logs indicate that limited radiological surveys were performed earlier in the outage, which demonstrated that radiation levels in the platform would not likely exceed acceptable levels. In this case, the workers were not exposed to any significant radiation levels. As corrective actions, the licensee posted information signs to control access to specific areas of the 360 platform and planned to incorporate more rigorous radiological controls into the governing procedure.

Analysis: The inspectors determined this finding was a performance deficiency because the licensee failed to implement procedurally required compensatory measures associated with moving irradiated fuel assemblies. Specifically, workers on the 360 platform were in close proximity (i.e., 5 feet) of the refueling mast with fuel moves in progress. The deficiency was reasonably within the licensee’s ability to foresee and correct. The finding was more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of protecting worker health and safety from exposure to radiation, in that not implementing adequate radiological control could potentially result in unplanned exposures to radioactive material. The finding was assessed using the Occupational Radiation Safety SDP and was determined to be of very low safety significance (Green) because it was not an ALARA planning issue, there was no overexposure or potential for overexposure, and the licensee's ability to assess dose was not compromised.

As described above, the licensee relied on limited radiological surveys, rather than using the guidance already established in station procedures. Consequently, the cause of this deficiency had a cross-cutting aspect in the area of human performance in accordance with IMC 0305 H.1(b). Specifically, the licensee failed to use conservative assumptions in decision making and to demonstrate that the proposed action is safe in order to proceed rather than a requirement to demonstrate that it is unsafe in order to disapprove the action

Enforcement: Technical Specification 5.4.1.a requires the licensee to establish, implement, and maintain adequate written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which include radiation safety and refueling procedures. Procedure SOI-F15, "Refueling and 360 Platforms," provides precautions and limitations for maintaining a required amount of water shielding. Step 2.33 states, "To ensure a minimum of 7'-1" of water shielding for personnel working in the troughs or underwater carriage, a minimum of 5 foot of horizontal separation between the refuel mast or auxiliary hoist load cable and personnel in the 360 Platform troughs and underwater carriage shall be maintained." Contrary to the above, in March 2009, the licensee failed to institute sufficient administrative controls to maintain a minimum 5-foot separation between the refuel mast or auxiliary hoist load cable and personnel in the 360 Platform troughs. Because the finding is of very low safety significance and has been entered in the licensee's CAP as CR 09-54697 this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000440/2009003-05)

.5 High Risk Significant, High Dose Rate, High Radiation Area and Very High Radiation Area Controls

a. Inspection Scope

The inspectors held discussions with the radiation protection manager concerning high dose rate, high radiation area and very high radiation area controls and procedures, including procedural changes that had occurred since the last inspection, in order to assess whether any procedure modifications substantially reduced the effectiveness and level of worker protection.

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

The inspectors discussed with radiation protection supervisors the controls that were in place for special areas of the plant that had the potential to become very high radiation areas during certain plant operations. The inspectors assessed if plant operations required communication beforehand with the radiation protection group, so as to allow corresponding timely actions to properly post and control the radiation hazards.

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

The inspectors observed several radiological briefs for worker entry into high radiation areas and conducted plant walkdowns to assess the posting and locking of entrances to elevated dose rate areas, high radiation areas, and very high radiation areas.

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

b. Findings

Introduction: A Green self-revealed finding of very low safety significance and associated NCV of TS 5.7.1.a was identified for the failure to barricade and conspicuously post a high radiation area on elevation 599' of the auxiliary building.

Description: On March 15, 2009, a worker traversing elevation 599' of the auxiliary building corridor reported a personal dosimeter dose-rate alarm to the radiation protection department. The area at the time of occurrence was posted as a radiation area. The radiation protection department initiated follow-up radiological surveys in response to the reported alarm and identified general area dose rates in the worker's travel path of up to 150 mrem/hr due to shine from suppression pool clean-up piping in the plant overhead. Some areas and sections of pipe were upwards of 2000 mrem/hr on contact and 600 mrem/hr at 30 cm. The worker received minimal radiation exposure from the incident.

Follow-up investigation by the radiation protection staff identified the original source of the elevated general area dose to be the draining and flushing of the reactor cavity. Specifically, the flow path for water and debris from reactor cavity clean-up activities was through the fuel pool clean-up system piping and through condensate storage and transfer system piping to the suppression pool clean-up system, with suppression pool clean-up pipes in the overhead of auxiliary building 599' being the source of the high radiation area conditions.

Water transfers from reactor cavity clean-up and decontamination work was a routinely planned refueling outage maintenance activity, with known and pre-defined system flow paths. Consequently, the radiological impact and the associated elevated dose rates in the immediate areas of affected system piping and components would not be unexpected. Immediate corrective actions included barricading and conspicuously posting the 599' auxiliary building corridor as a high radiation area. Additional actions included performing gravity flushes of piping with clean water to reduce the ambient dose rates. No additional radiological incidents of dosimeter dose or dose-rate alarms were identified as a result of this event.

Analysis: The inspectors determined that this finding was a performance deficiency because the licensee failed to meet the requirements contained in TS and because the deficiency was reasonably within the licensee's ability to foresee and correct. The finding was more than minor because it impacted the program and process attribute of the Occupational Radiation Safety cornerstone and affected the cornerstone objective of protecting worker health and safety from exposure to radiation, in that, not barricading and conspicuously posting high radiation areas could result in unnecessary and unplanned radiation exposures to workers. The finding was assessed using the Occupational Radiation Safety SDP and was determined to be of very low safety significance (Green) because it was not an ALARA planning issue, there was no overexposure or potential for overexposure, and the licensee's ability to assess dose was not compromised.

Additionally there is a cross-cutting aspect associated with this finding in the area of human performance in work control in accordance with IMC 0305 H.3(b). Specifically, the licensee failed to appropriately coordinate work activities by incorporating actions to address the impact of the work on different job activities and failed to maintain interdepartmental coordination necessary to assure plant and human performance.

Enforcement: Technical Specification 5.7.1 states, in part, that each high radiation area shall be barricaded and conspicuously posted. Contrary to the above, on March 15, 2009, the licensee failed to barricade and conspicuously post a high radiation area on elevation 599' of the auxiliary building. Since the failure to comply with the TS was of very low safety significance and has been entered in the licensee's CAP as CR 09-55453, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000440/2009003-06)

.6 Radiation Worker Performance

a. Inspection Scope

The inspectors reviewed radiological problem reports for which the cause of the event was due to radiation worker errors to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. Problems or issues with planned or completed corrective actions were discussed with the radiation protection manager.

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

b. Findings

No findings of significance were identified.

.7 Radiation Protection Technician Proficiency

a. Inspection Scope

The inspectors reviewed radiological problem reports for which the cause of the event was radiation protection technician error to determine if there was an observable pattern traceable to a similar cause and to determine if this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

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

b. Findings

Introduction: The inspectors identified an unresolved item (URI) concerning a contract employee (a radiation protection technician) who failed to appropriately respond to a portal monitor radiological contamination alarm when exiting the site.

Discussion: The inspectors reviewed a CAP record (CR 09-55585) that describes an issue dated March 15, 2009, in which a contract employee disregarded a portal monitor alarm in the plant access facility (PAF) when exiting the plant. Based on the initial assessment of the inspectors, there appears to be a potential performance deficiency associated with radioactive material control. The licensee's investigation into this issue was on-going. However, initial indications were that inadequate radiological contamination surveys were performed and that the radiation protection technician inappropriately responded to multiple radiological alarms of the portal monitor. The issue remains under review by the NRC to determine the performance deficiency and is categorized as URI 0500440/2009003-07.

2OS2 ALARA Planning And Controls (71121.02)

.1 Radiological Work Planning

a. Inspection Scope

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

This inspection constituted a partial sample as defined in IP 71121.02-5.

The inspectors evaluated the interfaces between operations, radiation protection, maintenance, maintenance planning, scheduling, and engineering groups to identify interface problems or missing program elements.

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

b. Findings

Introduction: The inspectors identified an URI associated with the licensee's ALARA planning and work controls for the alternate decay heat removal project.

Discussion: The inspectors reviewed a CAP record (CR 09-55801) that described the significant dose overrun on Radiation Work Permit (RWP) 09-6035, for the alternate decay heat removal project. The work was ongoing during the current refueling outage (RFO 12). Based on the initial assessment of the inspectors, there appeared to be a potential performance deficiency associated with the planning and execution of this work. The actual dose incurred was several times more than the initial dose estimate and the reasons and radiological impact for the increased dose was not well understood. Since the work activity was not completed, the inspectors could not assess the total dose for the activity nor the full scope and impact of the work controls. Consequently, this issue remains under review by the NRC to determine if it represents a performance deficiency and is categorized as URI 0500440/2009003-08.

.2 Job Site Inspections and ALARA Control

a. Inspection Scope

The inspectors reviewed exposures of individuals from selected work groups to evaluate any significant exposure variations among workers and to determine whether any significant exposure variations were the result of worker job skill differences or whether certain workers received higher doses because of poor ALARA work practices.

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

b. Findings

No findings of significance were identified.

.3 Source-Term Reduction and Control

a. Inspection Scope

The inspectors reviewed licensee records to evaluate the historical trends and the current status of tracked plant source terms. The inspectors determined if the licensee was making allowances and had developed contingency plans for expected changes in the source term due to changes in plant fuel performance issues or changes in plant primary chemistry.

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

The inspectors verified that the licensee had developed an understanding of the plant source term, including knowledge of input mechanisms to reduce the source term. The inspectors evaluated if the licensee had a source term control strategy in place that included a cobalt reduction strategy, shutdown controls, and operating chemistry plan, which was designed to minimize the source term external to the core. Other methods used by the licensee to control the source term including component and system decontamination and the use of shielding were also evaluated. Particular attention was given to the use of shielding associated with the RHR system in the auxiliary building and in the drywell. Additionally, the inspectors reviewed the planning, implementation, and results achieved by the chemical decontamination of the RCS and the associated radiological impact on ambient drywell dose rates.

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

The inspectors reviewed the licensee’s identification of specific sources of radiation, along with exposure reduction actions and the priorities the licensee had established for implementation of those actions. The results that had been achieved against these priorities since the last refueling cycle were reviewed. For the current assessment period, source reduction evaluations were verified along with actions taken to reduce the overall source term compared to the previous year.

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

b. Findings

No findings of significance were identified.

.4 Problem Identification and Resolutions

a. Inspection Scope

The inspectors verified that identified problems were entered into the CAP for resolution and that they had been properly characterized, prioritized, and resolved. The inspectors' review included dose significant post-job (work activity) reviews and post-outage ALARA report critiques of exposure performance.

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

The inspectors reviewed corrective action reports related to the ALARA program and interviewed staff members to verify that follow-up activities had been conducted in an effective and timely manner commensurate with their importance to safety and risk using the following criteria:

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

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

The inspectors reviewed the licensee's CAP to determine if repetitive deficiencies and/or significant individual deficiencies in problem identification and resolution had been addressed.

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

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

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

This inspection constituted one Unplanned Scrams per 7000 Critical Hours sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

.2 Unplanned Scrams with Complications

a. Inspection Scope

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

This inspection constituted one Unplanned Scrams with Complications sample as defined in IP 71151-05.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

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

.1 Routine Review of Items Entered Into the CAP

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

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

.1 Loss of Shutdown Cooling Due to Maintenance Activity

a. Inspection Scope

The inspectors reviewed the licensee's response to a loss of shutdown cooling (SDC) on April 27, 2009, when technicians who were installing jumpers inadvertently contacted an adjacent grounding bus bar, resulting in a blown fuse. This created an isolation signal that shutdown the RHR pump 'A', which was the operating SDC pump. Documents reviewed are listed in the Attachment.

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

b. Findings

Loss of Shutdown Cooling Water Flow to Reactor Vessel

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," was self-revealed when technicians performed maintenance on protected equipment without implementing risk management requirements specified in station procedures. This resulted in a loss of shutdown cooling flow to the reactor coolant system.

Description: On April 27, 2009, the plant was shut down in Mode 5 and the reactor refueling activities were completed. The reactor cavity was drained down to just below the RPV flange in preparation for RPV head installation. Residual heat removal pump (RHR) 'A' was providing cooling water flow to the RPV, and RHR 'B' was designated as the backup decay heat removal system. Time to boil was about 9 hours.

Technicians obtained the shift manager's permission to install jumpers in a control room electrical cabinet in preparation for an upcoming containment integrated leak rate test. This electrical cabinet was posted as protected equipment per the licensee's maintenance risk control program.

The procedure to install the jumpers was considered by the licensee to be an infrequently performed task. The lead technician was aware of operating experience related to tripping of the reactor protection system due to work in the affected cabinet. The work was considered challenging due to limited working space within the cabinet. While the technicians recognized that the task was challenging to perform without incident, the technicians did not share this information with their supervision or the shift manager. In addition, the technician's supervisor was unaware that work was to be performed on protected equipment and the licensee did not provide direct oversight for the technicians.

While installing jumpers in the cabinet, the lead technician mishandled the jumper lead twice allowing it to slip off of the terminal. Despite this, the technicians continued working without informing control room staff or their supervision of the difficulties. At about 5:30 p.m. on April 27, 2009, the technician slipped the jumper a third time and it contacted a ground bus bar within the cabinet. This caused a short circuit in the reactor protection system that resulted in a blown fuse and affected an RHR isolation logic circuit.

The common RHR suction isolation valve for shutdown cooling, 1E12F008, received an auto-isolation signal and closed. The closure of valve 1E12F008 tripped the 'A' RHR pump and prevented the 'B' RHR pump from being started. Shutdown cooling water flow to the RCS was lost. Operators replaced the blown fuse, opened 1E12F008, and started the 'B' RHR pump to restore shutdown cooling flow to the reactor. Shutdown cooling was restored at 6:35 p.m. on April 27, 2009, and the time that the reactor was without shutdown cooling was approximately 1 hour. During this time, indicated temperature at the reactor bottom head increased from about 94 °F to 97 °F.

Licensee procedure NOP-OP-1005, “Shutdown Defense in Depth,” Revision 10, Section 4.2.9 states, in part, “It is not desirable to perform maintenance or conduct testing on equipment which is being relied upon to provide necessary Shutdown Defense in Depth for the unit. If it is felt that the only possible recourse is to conduct testing or maintenance on one of these systems or components, then NOP-OP-1010, Operational Decision Making, will be utilized. Special emphasis, within the ODMI (Operational Decision Making Instruction), should be placed on contingency actions and oversight for these activities.” Contrary to the above, the licensee performed maintenance on protected equipment without implementation of an ODMI and without special emphasis on contingency actions and oversight.

Analysis: The inspectors determined that performing maintenance activities affecting shutdown cooling did not meet the requirements of station risk management procedure NOP-OP-1005, “Shutdown Defense in Depth,” Revision 10, and was a performance deficiency. The finding was more than minor because it was associated with the equipment performance attribute of the initiating events cornerstone and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the finding resulted in a loss of reactor decay heat removal event while the reactor was shut down.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, “Shutdown Operations Significant Determination Process.” The inspectors used Checklist 8 contained in Attachment 1 and determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance (Green) using Figure 1.

This finding has a cross-cutting aspect in the area of human performance, work control, per IMC 0305 H.3(a) because the licensee did not appropriately plan the work activity consistent with nuclear safety, incorporating risk insights, job site conditions, or the need for planned contingencies, compensatory actions, and abort criteria. Specifically, licensee personnel failed to recognize the risk significance of working in the protected cabinet, failed to address the tight-quarters work environment within the cabinet, and failed to appropriately plan contingency actions and abort criteria with respect to dropped leads and potential consequences.

Enforcement: Criterion V of 10 CFR Part 50, Appendix B, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. The licensee established procedure NOP-OP-1005, "Shutdown Defense in Depth," Revision 10 as the implementing procedure for risk management during shutdown conditions.

Contrary to the above, on April 27, 2009, the licensee failed to assess and manage the risk associated with maintenance affecting the residual heat removal system. Specifically, the licensee failed to implement the significant risk management actions prescribed in procedure NOP-OP-1005 for maintenance on protected equipment. Because this violation was of very low safety significance and it was entered into the licensee's CAP as CR 09-58110, this violation is being treated as a NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2009003-09)

.2 Near Loss of Shutdown Cooling Due to Maintenance Activity

a. Inspection Scope

The inspectors reviewed the licensee's response to a near loss of SDC on April 28, 2009, when technicians were testing non-safety relays associated with 15-kvolt bus L11 and main generator. The technicians worked the incorrect relay and tripped the relay. Due to a fault with the relay, it did not trip. If the relay had worked as designed, it would have resulted in the loss of the RPS motor generator set and closed the SDC inboard common isolation valve resulting in a loss of SDC.

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

b. Findings

Work On Wrong Relay Affecting Shutdown Cooling

Introduction: A finding of very low safety significance (Green) was self-revealed on April 28, 2009, for the failure to follow maintenance procedure PTI-N41-P0002, "Generator Switchgear Protective Relay Trip Test," when electricians performed maintenance on an incorrect relay associated with bus L11.

Description: On April 28, 2009, the plant was in Mode 5 with the reactor core refueled. Installation of the RPV head was in progress. The 'A' RHR pump was in service providing shutdown cooling water flow to the RCS. The 'B' train of RHR was designated as the backup decay heat removal system.

Electricians were performing procedure PTI-N41-P0002, "Generator Switchgear Protective Relay Trip Test," which required testing of relays associated with bus L11, a non-safety bus. Two electricians and a supervisor proceeded to switchgear L11 to perform procedure Section 5.1.20. The electricians had not previously walked down the job site and did not conduct a "2-minute drill." These error prevention techniques are licensee human performance management practice expectations, required upon arrival at the work site.

The first step of Section 5.1.20 stated, "At L1102 momentarily close the trip contact of relay 1R22-Q103D, the A phase over-current relay." The electricians verified they were working on the L1102 bus cubicle and proceeded to verify the relay to be tested. The electricians selected the relay labeled "51 PH A" to work. The correct relay was labeled "51 PH A CT ON AUX XFMR." The electricians closed the trip contact on the relay labeled "51 PH A." This relay is the over-current relay for the generator lock-out circuitry, 1R22-Q103A. Relay 1R22-Q103A began to smoke and the electricians pulled the relay from the circuit and de-energized the relay. Bus relay 86B, the lockout relay for bus L11, failed to actuate when 1R22-Q103A was de-energized. Had the relays functioned as designed, power would have been lost to the reactor protection system circuitry resulting in an isolation signal being sent to valve 1E12F008, the common suction isolation valve for shutdown cooling. Therefore, had 1R22-Q103A and 86B relays functioned properly, RHR valve 1E12F008 would have closed, tripping the RHR 'A' pump and preventing the start of the RHR 'B' pump resulting in the loss of shutdown cooling to the reactor.

The licensee entered the issue into their CAP. The licensee posted bus L11 as a protected train and repaired the 1R22-Q103A and 86B circuitry.

Analysis: The inspectors determined that the maintenance activities related to relay 1R22-Q103A were not performed in accordance with procedure PTI-N41-P0002, "Generator Switchgear Protective Relay Trip Test, and was a performance deficiency. The finding was determined to be more than minor because the finding was associated with the Initiating Events cornerstone attribute of equipment performance and affected the cornerstone objective of limiting the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, had the 1R22-Q103A relay circuitry functioned as designed, a loss of decay heat removal event would have occurred.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, "Shutdown Operations SDP." The inspectors used Checklist 8 contained in Attachment 1 and determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance (Green) using Figure 1.

This finding has a cross-cutting aspect in the area of human performance, work practices, per IMC 0305 H.4(a) because the licensee did not use error prevention techniques commensurate with the risk of the maintenance activity. Specifically, the electricians did not perform appropriate self and peer checking when faced with similarly labeled relays on the panel to be worked, electricians did not walk down the job site prior to commencing work, and the supervisor in the field had not attended the pre-job brief.

Enforcement: Enforcement action does not apply because the performance deficiency did not involve a violation of regulatory requirements. Because the finding does not involve a violation of regulatory requirements, is of very low safety significance, and the issue was addressed in the CAP as CR-09-58187, it is identified as (FIN) 05000440/2009003-10.

.3 Over-pressurization of the RHR 'A' System During Reactor Vessel Hydrostatic Pressure Test

a. Inspection Scope

The inspectors reviewed the licensee's response to the over pressurization of the RHR 'A' system on May 7, 2009, during RCS leakage pressure test. The RHR 'A' system pressure increased to greater than system design pressure, 500 psi, due to a leaking check valve, 1E12-F0050A, when the RCS pressure was raised to 1030 psi.

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

b. Findings

RHR System Over-Pressurization Due to Failure to Implement Corrective Actions for a Condition Adverse to Quality

Introduction: A finding of very low safety significance (Green) and associated NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," was self-revealed for the failure to establish and maintain corrective actions to ensure RHR check valve 1E12F0050A seated during plant pressurizations.

Description: On May 7, 2009, during performance of the RCS leakage test, the RHR 'A' system was isolated by the means of an isolation valve and check valve 1E12F0050A. The RCS pressure was between 1025 to 1050 psig. The RHR 'A' system design pressure was 500 psig at 358 °F. The RCS temperature was approximately 150 °F. At 11:26 a.m. the RHR 'A' system high pressure alarm was received and operators noted the system pressure indication was above 500 psig. The pressure instrument indication range was 0 to 500 psig, and the operators were unable to determine actual RHR system pressure.

The operators took manual action to depressurize the RHR 'A' system to less than 500 psig and continued to monitor system pressure. System pressure continued to rise and the operators had to manually depressurize the system at 11:53 a.m., 12:41 p.m., 1:34 p.m., and 2:20 p.m. to maintain RHR pressure at approximately 400 psig. The licensee's analysis determined that at 200 °F RHR system pressure limit would be 700 psig. The licensee's calculations determined that with two protective relief valves and the observed rate of change of pressure increase, that the maximum RHR system pressure would be less than 700 psig.

The licensee's investigation determined that leak by of the check valve 1E12F0050A had occurred previously. The licensee implemented procedures to vent a section of piping when increasing RCS pressure to ensure the check valve would seat properly. In July 2007 operations personnel requested to remove this requirement in CR 07-24161 due to personnel safety concerns. The system operating instruction (SOI) and integrated operating instruction (IOI) were revised to remove this evolution. The licensee did specify the replacement of the check valve in the next refueling outage, but this activity was not included in the outage work scope. After this decision the licensee did not specify an alternate corrective action for this issue. The licensee entered this issue into the CAP and is evaluating the appropriate course of action.

Analysis: The inspectors determined that the failure to implement corrective actions to ensure the check valve seated during plant pressurizations was contrary to 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," The inspectors determined that the finding was more than minor because it was associated with the procedure quality attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, removal of the RHR venting evolution from station procedures resulted in an unexpected over-pressurization which could have resulted in system damage.

The inspectors determined that the finding could be evaluated in accordance with IMC 0609, Appendix G, "Shutdown Operations Significant Determination Process." The inspectors used Checklist 8 contained in Attachment 1 and determined that the finding did not require a Phase 2 or Phase 3 analysis because the plant had appropriately met the safety function guidelines for core heat removal, inventory control, power availability, containment integrity, and reactivity control. The issue did not need a quantitative assessment and screened as having very low safety significance (Green) using Figure 1.

This finding has a cross-cutting aspect in the area of problem identification and resolution per IMC 0305 P.1(c), because the organization failed to thoroughly evaluate the impact of modifying a corrective action. Specifically, the licensee failed to thoroughly evaluate the consequences of removing the venting section of a procedure that was a corrective action for the check valve's inability to seat under low differential pressure conditions.

Enforcement: Criterion XVI of 10 CFR Part 50, Appendix B, "Corrective Action," requires, in part, that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected.

Contrary to the above, since July 2007 the licensee failed to promptly identify and correct a condition adverse to quality regarding an RHR check valve failing to seat during RCS pressurization. Specifically, corrective actions to address check valve 1E12F0050A failure to seat had been discontinued without implementing alternate corrective actions.

Because this violation was of very low safety significance and was entered into the licensee's CAP as CR 09-58808 and CR 09-58995, this violation is being treated as an NCV, consistent with Section VI.A.1 of the NRC Enforcement Policy. (NCV 05000440/2009003-11)

.4 Trip of the 'A' Reactor Recirculation Pump When Attempting to Start it in Fast Speed

a. Inspection Scope

The inspectors reviewed the licensee's response to the loss of the 'A' reactor recirculation (RR) pump on May 14, 2009, at 10:52 p.m. The licensee was attempting to shift the 'A' RR pump from slow to fast speed when the breaker opened on the instantaneous overload current relay. After exploring all potential causes for the breaker trip, the licensee could not determine the cause for the relay activation. The licensee returned the pump to service on May 16, 2009, and subsequently shifted it to fast speed with no incident. The licensee continues to monitor the pump.

b. Findings

No findings of significance were identified.

.5 Main Turbine Trip and Reactor Scram

b. Inspection Scope

The inspectors reviewed the licensee's response to a main turbine trip and subsequent reactor scram. The inspectors evaluated the initiating cause of reactor scram and the personnel response requiring more than routine operator actions. The inspectors determine that the response was appropriate and in accordance with procedures and training. The inspectors reviewed operator logs; plant computer data, strip charts, and other data after stable plant conditions were achieved. The inspectors also monitored plant startup activities.

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Licensee Activities and Meetings

The inspectors observed select portions of licensee activities and meetings and met with licensee personnel to discuss various topics. The activities that were sampled included:

.2 Quarterly Resident Inspector Observations of Security Personnel and Activities

a. Inspection Scope

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

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

b. Findings

No findings of significance were identified.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. M. Bezilla, Site Vice President and other members of licensee management on July 15, 2009. A supplemental exit was conducted on July 23, 2009, to discuss a change in the cross-cutting issue associated with item 05000440/2009002-11 (Section 4OA3.3). The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meetings

Interim exits were conducted for:

- The preliminary results of the licensee's access control to radiologically-significant areas, and the ALARA planning and controls program with the Site Vice President, Mr. M. Bezilla, on April 03, 2009.
- The results of the triennial heat sink performance inspection were presented to Mr. K. Krueger, Plant General Manager; and other members of the licensee's staff on Friday, June 5, 2009.

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

4OA7 Licensee-Identified Violations

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

- Technical Specification 5.4.1.a requires the licensee to establish, implement, and maintain adequate written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which includes procedures for a radiation work permit system. Procedure NOP-OP-4107, "Radiation Work Permits," Section 4.4.2 defines radiological briefs. Contrary to the above, on March 30, 2009, a station worker entered a locked high radiation area without the required radiological brief, and radiological conditions had changed since the time of the worker's last area entry. This was identified in the licensee's CAP as CR 09-56298. Corrective actions included briefing the individual on the radiological work environment and performance management of the individuals involved in accordance with station protocol. The finding was determined to be of very low safety significance because it was not an ALARA planning issue, there was no overexposure nor potential for overexposure, and the licensee's ability to assess dose was not compromised.
- Technical Specification 5.4.1.a requires the licensee to establish, implement, and maintain adequate written procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978, which includes procedures for control of radioactive material. Procedure HPI-L004, "Source Control Documentation and Inventory," Section 4.6 defines the issuance and return of radioactive sources.

Contrary to the above, on February 27, 2009, an individual left site in possession of an exempt quantity Cesium-137 instrument check source. The source was not logged out as being in use, as was required by station procedures. Under less favorable conditions, traceability of source use, location, and possession would have been much more complicated. This issue was identified in the licensee's CAP as CR 09-54406. Corrective actions included retuning the source to the station's control and performance management of the individual involved in accordance with station protocol. The finding was determined to be of very low safety significance because it was a radioactive material control issue, not involved with transportation and the public exposure was less the 0.005rem.

- Appendix B, Criterion III, "Design Control", of 10 CFR 50, states, in part, that measures shall be established to assure that applicable regulatory requirements and design basis, as defined in 50.2 and as specified in the license application, for those SSCs to which this appendix applies are correctly translated into specifications, drawing, procedures, and instructions. Contrary to the above, CR-09-56310 documents that the original specifications for ECCS waterleg pumps used non-conservative seismic response spectra for vendor qualification. This latent design issue is a nonconforming condition that affects the potential dynamic amplification of ground acceleration. Subsequent calculations using actual installed data shows that sufficient seismic margin exists to support continued operability of all ECCS waterleg pumps. The issue was entered into the CAP as CR-0-56310.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Bezilla, Vice President Nuclear
K. Krueger, Plant General Manager
A. Cayia, Director, Performance Improvement
K. Cimorelli, Director, Maintenance
A. Duszynski, Generic Letter 89.13 Program Owner
D. Evans, Manager, Operations
J. Grabner, Director, Site Engineering
E. Gordon, Radiation Protection Superintendent
H. Hanson, Jr., Director, Work and Outage Management
P. McNulty, Radiation Protection Manager
P. New, Radiation Protection
J. Pelcic, Nuclear Compliance
A. Pusateri, Supervisor of Plant Balance for Systems Engineers

NRC

S. Burgess
D. Passehl

LIST OF ITEMS OPENED, CLOSED, DISCUSSED

Opened and Closed

05000440/2009003-01	NCV	RCS Temperature Below Minimum Allowed by TS Due to Inadequate Station Procedures (Section 1R15.1)
05000440/2009003-02	FIN	Failure to Follow Industrial Safety Manual Results in Damage to Fuel Handling Building Roll-up Door (Section 1R15.2)
05000440/2009003-03	NCV	Inability to Operate the RHR Common Suction Line Valve (Section 1R20)
05000440/2009003-04	NCV	Failure to Prevent Contact of Energized Components Renders RCIC System Inoperable (Section 1R22)
05000440/2009003-05	NCV	Failure to Implement Adequate Compensatory Measures Associated with Moving Burned Fuel Coincident with Workers on the 360 Platform (Section 2OS1.4)
05000440/2009003-06	NCV	Failure to Barricade and Conspicuously Post a High Radiation Area. (Section 2OS1.5)
05000440/2009003-09	NCV	Loss of Shutdown Cooling Water Flow to Reactor Vessel (Section 4OA3.1)
05000440/2009003-10	FIN	Work on Wrong Relay Affecting Shutdown Cooling (Section 4OA3.2)
05000440/2009003-11	NCV	RHR System Over-pressurization Due to Failure to Implement Corrective Actions for a Condition Adverse to Quality (Section 4OA3.3)

Opened

05000440/2009003-07	URI	Employee Disregarded Portal Monitor Alarm. (Section 2OS1.6)
05000440/2009003-08	URI	Dose Overrun on RWP 09-6035 for the Alternate Decay Heat Removal Project (Section 2OS2.1)

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather

ONI-R10; Loss of AC Power; Revision 9
ONI-S11; Hi/Low Voltage; Revision 5
NOP-OP-1003; Grid Reliability Protocol; Revision 2
PAP-0102; Interface with the Transmission System Operator; Revision 5
CR 08-44379, PTI-M99P0001 Ambient Temperature Monitoring Summer Performance, dated August 6, 2008
CR 07-20483, Summer Prep Orders Rescheduled Due to Unresolved Restraints
NOP-WM-2001, Work Management Scheduling/Assessment/Seasonal Readiness Process, Revision 7
IOI-15, Seasonal Variations, Revision 16

1R04 Equipment Alignment

VLI-E21, Low Pressure Core Spray System, Revision 8
VLI-R45, Division 1 and 2 Diesel Generator Fuel Oil System (Unit 1), Revision 5
VLI-R44, Division 1 and 2 Diesel Generator Starting Air System (Unit 1), Revision 4
VLI-R46, Division 1 and 2 Diesel Generator Jacket Water System (Unit 1), Revision 4
VLI-M25/26, Control Room HVAC and Emergency Recirculation System, Revision 7
SOI-M25/26, Control Room HVAC and Emergency Recirculation System, Revision 17

1R05 Fire Protection (Annual/Quarterly)

CR 09-58050; Follow-up on Drywell CO2 Alarm; dated April 27, 2009
CR 09-58507; Fire/Tornado Door CC-323 Found 'Stuck' Full Open; dated May 3, 2009
CR 09-58651; Fire Door HB-304 Inoperable; dated May 5, 2009
CR 09-58641; Fire Door Not Auto-Closing; dated May 5, 2009
CR 09-58530; IsoPhase Bus Deluge Will Not Build Priming Pressure/Failed PTI-P54-P0063; dated May 3, 2009
CR 09-58702; Aux Transformer Deluge Would Not Reset; dated May 4, 2009
CR 09-58703; Multiple Alarms Received on Fire Panel; dated May 6, 2009
FPI-OCC, Control Complex, Revision 7
FPI-A-A02, Periodic Fire Inspections, Revision 5
PAP-1910, Fire Protection Program, Revision 18
PAP-0204, Housekeeping/Cleanliness Control Program, Revision 22

1R06 Internal Flooding

CR 09-59279, Drywell Floor Drain Cover Has Machined Rectangular Hole, dated May 10, 2009
CR 09-58868, PY-PTI-G61P0001 Drywell Floor Drain Sump Flow Capacity Test Results, dated May 8, 2009
CR 09-55755, Steam Tunnel Floor Drain Backing Up, dated March 20, 2009

1R07 Triennial Heat Sink

P42-039-Design Basis Heat Load and Required ESW Flow for the ECC HXs; Revision 2
P42-050-Emergency Closed Cooling Heat Exchanger "A" Loop Performance Test Evaluation; Revision 1
P-45-44-Keepfill Check Valve Leak Rate/Standpipe Drain-down Level; Revision 2
P45-085-Design Function of ESW Pump Discharge Valves and Associated Leakage Criteria; Revision 0
ISI-GEN-T3000-EC-060 Pipe Wall Thickness Monitoring Examination Sign-off Sheet; dated November 14, 2008
ISI-GEN-T3000-EC-071 Pipe Wall Thickness Monitoring Examination Sign-off Sheet; dated December 15, 2008
EMARP-0008-ESW 'C' Discharge Strainer; dated March 26, 2009
EMARP-0008-1O45D0002A ESW 'A' Strainer; dated April 15, 2009
EMARP-00011-ESW Pump House Structural Inspection Report; dated November 19, 2007
EMARP-0011-ESW Pump House Forebay, Intake Traveling Screens and Normal Intake Tunnel Riser Outlet, dated April 21, 2008
CR 09-59963-A typographical error on calculation P42-050; Revision 1; dated June 3, 2009
CR 09-60078-Extension ladder in RHR Pump Room "B" not adequately secured; dated June 3, 2009
CR 09-60149-Valve Packing Leak; dated June 2, 2009
1P42B0001A-Emergency Closed Cooling Heat Exchanger; dated January 19, 2006
302-0793-00000-Emergency Service Water Operating Data; Revision N
ARI-H13-P601-0020-G4-Sluice Gate A/B Open – ESW Forebay Low; Revision 14
CHI-0004-System Chemical Treatment; Revision 13
EMARP-0008-Clam/Mussel Monitoring; Revision 7
EMARP-011-Emergency Service Water System Monitoring Program; Revision 5
IOI-15-Seasonal Variations; Revision 16
NOP-OP-3602-Microbiologically Influenced Corrosion Monitoring Program; Revision 00
PTI-GEN-P0024-Mussel Treatment; Revision 15
REC-0104-Chemistry Specifications Attachment 25: Closed Cooling and Heating System-30
REC-0104-Chemistry Specifications Attachment 38: Emergency Service Water; Revision 30
SOI-P48/P84B-Service Water and Emergency Service Water Chlorination and Dechlorination System; Revision 17
WO 960005554-ECC Train "A" Heat Exchanger Inspection Report (visual and cleaning); dated October 2, 1997
WO 200068834-ECC Train "A" Heat Exchanger Inspection Report (visual and cleaning); dated March 2, 2005
WO 200144120-ECC Train "A" plugging of Heat Exchanger leaking tube; dated March 26, 2005
WO 200269350-ESW Forebay Low Level Functional For P45-D004A; dated January 14, 2009
WO 200269351-ESW Forebay Low Level Functional For P45-D004B; dated January 14, 2009
WO 200272759-ESW System Loop B Flow and Differential Pressure Test; dated January 1, 2009
WO 200272764-ESW System Loop A Flow and Differential Pressure Test; dated February 19, 2009
WO 200272857-ESW Pump A and Valve Operability Test; dated January 23, 2009
WO 200279942-ESW Pump A and Valve Operability Test; dated April 4, 2009
WO 200321690 -PY-P45 Emergency Service Water; dated April 28, 2009

1R11 Licensed Operator Requalification Program

Simulator Guide OTLC-3058200902_PY-SGC2, Revision 0, dated May 13, 2009
Simulator Examination Summary Sheet, dated May 29, 2009

Crew Competency Grading Worksheet, dated May 29, 2009
Overall Dynamic Simulator Individual Evaluation (SRO), dated May 29, 2009
Overall Dynamic Simulator Individual Evaluation (RO), dated May 29, 2009

1R12 Maintenance Effectiveness
Perry Nuclear Power Plant Health Report 2008-04
CR 09-54333; Screws Missing From Turbine Lube Oil Bearing Strainer; dated February 24, 2009
CR 09-53535; Lube Oil Storage Room Selector Valve Pilot; dated February 13, 2009
CR 09-52015; Lube Oil Purifier Room CO2 Control Panel; dated January 15, 2009
CR 08-49558; Possible Leak Main Turbine Lube Oil; dated November 16, 2008
CR 08-36909; Evaluate Main Turbine Lube Oil Usage; dated March 16, 2008
CR 09-60866; Potential Unrecognized Entry into LCO; dated June 22, 2009
CR 09-60493; Diesel Generator CO2 Panel HVAC Wiring Cross Landed Between Divisions; dated June 12, 2009
CR 09-61006; DIV 3 DG CO2 System to Failed to Trip Fans and Align Dampers per PTI P54 P0034C; dated June 24, 2009
WO 200194345 Standby Diesel Generator Div 2 CO2 Control Panel; dated November 01, 2008

1R13 Maintenance Risk Assessments and Emergent Work Control
CR 09-58318; Re-Evaluate Work Activities Allowed within Protected Equipment Postings; dated April 30, 2009
CR 09-58749; Inattention to Details While Processing Tagout for HCU 06-47; dated May 6, 2009
Perry Refuel 12 Outage Schedule
Perry Refuel 12 Power Ascension Schedule
CR 09-60006; Discovered Water/Foam in Diesel Fire Pump Room; dated July 3, 2009
PSA Risk Assessment; Period 9 Week 06; Revision 2
PSA Risk Assessment; Period 9 Week 08; Revision 0
PSA Risk Assessment; Period 9 Week 09; Revision 1

1R15 Operability Evaluations
CR 09-55397; Reactor Bottom Drain Temperature Lowered to Less Than 70 °F; dated March 14, 20009
CR 09-55268; IOI-9/TS 3.4.11 RPV Temperature Restrictions; dated March 12, 2009
IOI-9; Refueling; Revision 23
Evaluation of Reactor Vessel Temperature Excursion Below 70 Degrees F, CR 09-55397-03; dated March 24, 2009
First Energy Nuclear Operating Company Industrial Safety Manual; effective Aug 31, 2008
CR 09-60483; Excessive Fluid Coming From RB#5 During Div 1 DG Prestart Inspection, dated June 12, 2009
Cooper Enterprise Service Information Memo #409; Lube Oil Check Valve Rework R&RV Engines; Revision 0
CR 09-60561, Div 1 DG Oil In Power Cylinder, dated June 15, 2009
NOP-OP-1009, Operability Determinations and Functionality Assessments, Revision 1
WO 200004061, Unit 1-Division 1 and 2 Standby Diesel Generators, Predictive Maintenance/Rebuild/Replace Check Valves, dated June 17, 2009
DWG 302-0353-00000, Standby Diesel Generator Lube Oil, Revision S

1R19 Post-Maintenance Testing

WO 200262657; Post-Accident Sampling System (Pass) 'B' RHR Sample Functional Pressure Test – Class 4; dated April 27, 2009
WO 200258567; Residual Heat Removal Shutdown Cooling Suction Piping (E12) System In-service Pressure Test – Class 2; dated April 27, 2009
WO 200268221; Post-Accident Sampling System Valves Remote Position Indication Verification (P1); dated April 28, 2009
WO 200261803; Division II Shutdown Cooling Check Valve Exercise Test; dated April 27, 2009
WO 200328844; Master Trip Unit: Scram Discharge Volume Preventative Replacement; dated May 29, 2009
WO 200329474; Control Room HVAC Troubleshoot/Repair; dated April 10, 2009

1R20 Outage Activities

CR 09-56938; 1E12F0010 Improper Valve Indication; dated April 8, 2009
CR 09-57011; Shutdown Cooling Isolation Valve Has Abnormal Indication; dated April 9, 2009
CR 09-57119; LPCS Pump Ran on Minimum Flow for Greater than 1 Hour; dated April 11, 2009
CR 09-57053; Mechanical Seal Leak on RHR A Pump; dated April 10, 2009
CR 09-57128; Snubber Out of Tolerance; dated April 10, 2009
CR 09-57124; Vibration Data in Alert Range RHR A; dated April 12, 2009
CR 09-57123; Measured Vibration Amplitude Exceeds Set Schedule Maintenance Limits; dated April 12, 2009
CR 09-57146; DW Floor Drain Sump Pump A PMT Unsat; dated April 13, 2009
CR 09-57175; Wrong Blank Flange Removed per Order 200332135 in Containment; dated April 14, 2009
CR 09-57288; Recirc Motor Cannot be Aligned to Recirc Pump B within Acceptable Criteria; dated April 15, 2009
CR 09-57354; Ran RHR Pumps B & C on Min Flow for Greater Than One Hour During SVI-R43-T5367; dated April 16, 2009
CR 09-57509; Emergency Service Water (ESW) 18" Gate Valve; dated April 20, 2009
CR 09-57515; Clearance Boundary Red Tagged Valve Removed from System; dated April 20, 2009
CR 09-57523; Lower Airlock Door Maintenance Problems; dated April 20, 2009
CR 09-57545; Reactor Core Isolation Cooling Turbine Component Dimensions Out of Tolerance; dated April 20, 2009
CR 09-57542; Inconsistent Use of the Design Change Process in RFO12 Relative to Backfill; dated April 2, 2009
CR 09-57618; ECP 09-0150-001 Required Rework; dated April 21, 2009
CR 09-57702; Shutdown Safety Protected Equipment Posting Negated; dated April 22, 2009
CR 09-57903; 1E12F0010 Stem Would Not Align During Reassembly; dated April 23, 2009
CR 09-57964; RHR B Discharge Pressure Low; dated April 25, 2009
CR 09-57980; Dropped SHAZAM Bolt Tool; dated April 26, 2009
CR 09-58089; Poor Housekeeping Control of ESW Scaffold Storage Area; dated April 27, 2009
CR 09-58096; RHR A/LPCS Waterleg Pump Seal Leakage; dated April 27, 2009
CR 09-58138; Clearance Program Trending; dated April 28, 2009
CR 09-58153; Firewatch Not Completed in the Required Time; dated April 28, 2009
CR 09-58123; Tech Spec Action to be Met During Loss of S/D Cooling; dated April 27, 2009
CR 09-58318; Re-evaluate Work Activities Allowed Within Protected Equipment Postings; dated April 30, 2009
CR 09-58452; Partially Secured Grating in the Drywell and LPCS Pump Room; dated April 25, 20009
CR 09-58503; Steam Tunnel Hatch Plugs Installed Reverse of Drawings; dated May 2, 2009

CR 09-58526; 1P45F0068A Valve Indication Drifts Intermediate with ESW A In Operation; dated May 3, 2009
CR 09-58479; Foreign Material Floating in the Suppression Pool; dated May 2, 2009
CR 09-58494; NCC Low Flow to Recirc Pump B Upper Bearing did not Actuate the Annuciator; dated May 2, 2009
CR 09-58684; Unplanned Tech Spec Entry for Source Range Monitor Instruments; dated May 1, 2009
CR 09-58424; NPS Welder Issued and Used Weld Metal After SMAW Qualifications Expire; dated May 1, 2009
CR 09-58466; Hydrogen Igniter Failure 1M56S0102; dated May 2, 2009
CR 09-58594; Drywell/Containment Restart Inspection Observation; dated May 4, 2009
CR 09-58454; Issues Indentified During Maintenance Rule Walkdown of Unit 1 Annulus; dated April 30, 2009
CR 09-58601; Potential Mispositioning Outer Knob of Calibration Select/Command Switch; dated May 4, 2009
CR 09-58749; Inattention to Details While Processing Tagout for HCU 06-47; dated May 6, 2009
CR 09-58790; TIP E Failing Cold Crank Torques; dated May 7, 2009
CR 09-58804; RPV VT2 Not Coordinated with Valve Torque/Leak Inspections; dated May 7, 2009
CR 09-58855; Drywell Airlock Seal Pneumatic System Leak Test Unsat; dated May 8, 2009
CR 09-58758; Drywell Airlock Failed Pneumatic Seal Decay Test; dated May 5, 2009
CR 09-58845; Reactor Head Thermocouple Not Reading Correctly; dated May 7, 2009
CR 09-58961; Control Rod 34-43 Temperature Indication Lost; dated May 5, 2009
CR 09-58820; Non-ISI Leaks Identified During Performance of ISI-C11-T1102-2 on CRD HCU's; dated May 7, 2009
CR 09-58877; RWCU High Suction Flow With System Shutdown; dated May 8, 2009
CR 09-58903; SRM A Inop Due To Elevated Counts and Spiking; dated May 8, 2009
CR 09-58923; TXI-0377 and TXI-0378 Do Not Meet Requirements of PAP-1107; dated May 9, 2009
CR 09-59098; 1GG33F0038 5 DPM Leak-Reworked Pipe Cap Under Order 200369681; dated May 12, 2009
CR 09-59108; 1G33F0037 and 1G33F0038 Have Bent Packing Studs; dated May 13, 2009
CR 09-59115; Mispositioned Valve; dated May 13, 2009
CR 09-58814; Removal of Info Tags Prior to Approval Signature; dated May 7, 2009
CR 09-58800; Documentation of Control Rods Requiring Alternate Methods to Move; dated May 7, 2009
CR 09-58914; Safety-Related Air Issues; dated May 7, 2009
CR 09-59114; Elevated Temperature in the 12B Charcoal Absorber Bed; dated May 13, 2009
CR 09-59115; Mispositioned Valve; dated May 13, 2009
NOP-1014; Plant Status Control; Revision 00

<u>1R22 Surveillance Testing</u>
ISI-B21-T1300-1, Rev 15; Reactor Coolant System Leakage Pressure Test
SVI R43-T1337; Division 1 Standby Diesel Generator Loss Of Offsite Power (LOOP) Test; Revision 14
SVI R43-T1338; Division 2 Standby Diesel Generator Loss Of Offsite Power (LOOP) Test; Revision 18
SVI E22-T5397; HPCS Initiation and Loss of EH13 Response Time Test; Revision 11

SVI-E51-T2001; RCIC Pump and Valve Operability Test; Revision 31
WO 200289925; RCIC Steam Line Flow High Channel Functional for 1E31-N684B; dated May 18, 2009
SVI-E31- T5395B, "Reactor Core Isolation Cooling (RCIC) Steam Line High Channel Functional For 1E31-N0684-B," Revision 2
NOP-WM-4006; Conduct of Maintenance; Revision 2
CR 09-59356; Inadvertent Isolation of Reactor Core Isolation Cooling System; dated May 18, 2009
CR 09-59401; RCIC Trip Throttle Valve Tripped with No Steam Flow; dated May 18, 2009

2OS1 Access Control to Radiologically Significant Areas

CR 09-54406; Individual Took a Radioactive Source Off-Site to Their Residence; dated February 27, 2009
CR 09-54697; Compliance with SOI-F15 Precaution and Limitation 2.33; dated March 4, 2009
CR 09-56298; LHRA Entry without Re-Brief, RWP Violation; dated March 30, 2009
ECP No. 05-0186; Replacement of Auxiliary Platform IF15E005; Revision 0
HPI-B0003; Processing of Personnel Dosimetry; Revision 23
HPI-C0014; Radlock Key Issue; Revision 01
HPI-L004; Control of Radiography Operations; Revision 03
HPI-L004; Source Control Documentation and Inventory; Revision 06
IOI-9; Refueling; Revision 18
NOP-OP-4101; Access Control for Radiologically Controlled Areas; Revision 01
NOP-OP-4107; Radiation Work Permits; Revision 03
ONI-D17; High Radiation Levels within Plant; Revision 15
PAP-0114; Radiation Protection Program; Revision 14
PDB-R0001; Operational Requirements Manual; Revision 24
PTI-F15-P0001; Refueling Platform; Revision 09
RPI-0504; Radiologically Restricted Area Diving Program; Revision 06
RPI-0122; Temporary Shielding Program; Revision 05
SOI-F15; Refueling and 360 Platforms; Revisions 13 and 14
SVI-F15-T1349; Refueling Platform Operability Test; Revision 10

2OS2 As-Low-As-Is-Reasonably-Achievable Planning and Controls
Radiation Work Permit and Associated ALARA Files; RWP 096019; RFO-12 Refueling Activities; Revision 0
Radiation Work Permit and Associated ALARA Files; RWP 096020; RFO-12 In-Vessel Inspection (IVVI) Activities; Revision 0
Radiation Work Permit and Associated ALARA Files; RWP 096021; RFO-12 Radiography Activities; Revision 0
Radiation Work Permit and Associated ALARA Files; RWP 096035; RFO-12 Alternate Decay Heat Removal Project; Revision 0
Radiation Work Permit and Associated ALARA Files; RWP 096040; RFO-12 Suppression Pool Diving Activities; Revision 0
Radiation Work Permit and Associated ALARA Files; RWP 096042; RFO-12 RHR Motor Replacement; Revision 0
RWP Dose Estimate Tracking and Worksheets; Various dates 2009

4OA1 Performance Indicator Verification
NOBP-LP-4012-01, Rev 1; Unplanned Scrams per 7,000 Critical Hours; April 2008
NOBP-LP-4012-01, Rev 1; Unplanned Scrams per 7,000 Critical Hours; May 2008

NOBP-LP-4012-01, Rev 1; Unplanned Scrams per 7,000 Critical Hours; June 2008
NOBP-LP-4012-01, Rev 1; Unplanned Scrams per 7,000 Critical Hours; July 2008
NOBP-LP-4012-01, Rev 1; Unplanned Scrams per 7,000 Critical Hours; August 2008
NOBP-LP-4012-01, Rev 2; Unplanned Scrams per 7,000 Critical Hours; September 2008
NOBP-LP-4012-01, Rev 2; Unplanned Scrams per 7,000 Critical Hours; October 2008
NOBP-LP-4012-01, Rev 2; Unplanned Scrams per 7,000 Critical Hours; November 2008
NOBP-LP-4012-01, Rev 2; Unplanned Scrams per 7,000 Critical Hours; December 2008
NOBP-LP-4012-01, Rev 2; Unplanned Scrams per 7,000 Critical Hours; January 2009
NOBP-LP-4012-01, Rev 2; Unplanned Scrams per 7,000 Critical Hours; February 2009
NOBP-LP-4012-01, Rev 2; Unplanned Scrams per 7,000 Critical Hours; March 2009
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; April 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; May 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; June 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; July 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; August 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; September 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; October 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; November 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; December 2008
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; January 2009
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; February 2009
NOBP-LP-4012-02, Rev 3; Unplanned Scrams with Complications; March 2009

<u>4OA2 Identification and Resolution of Problems</u>
CR 09-59225; Cross-Cutting Theme for PI&R Aspect P.1(c), Problem Evaluation; dated May 14, 2009
CR 09-59237; Two Safety Significant Condition Reports Classified 'AC', No Investigation; dated May 14, 2009

<u>4OA3 Followup of Events and Notices of Enforcement Discretion</u>
CR 09-58110; Loss of Shutdown Cooling; dated April 27, 2009
NOP-OP-1005; Shutdown Defense in Depth; Revision 10
PAP-1924; Risk Informed Safety Assessment and Risk Management; Revision 4
PTI-N41-P0002; Generator Switchgear Protective Relay Trip Test; Revision 9
CR 09-58123; Tech Spec Action Unable to be Met During Loss of S/D Cooling; dated April 27, 2009
CR 09-58808; RHR A High Pressure During ISI-B21-T1300-1; dated May 7, 2009
CR 07-31207; E12-F0050A Failed to Seat; dated December 8, 2007
CR 07-24161; Missed Performance of SOI-E12 Section 7.13.5, IOI-1; dated July 26, 2007
CR 09-58995; Procedure Change to Vent Piping Between RHR to FW Return Isolation Valves; dated May 8, 2009
CR 09-60843; Condensate Minimum Flow Valve Did Not Open Following Reactor Scram; dated June 21, 2009
CR 09-60847; Motor Feed Pump Breaker L1007 Has Tripped Target; dated June 22, 2009
CR 09-60850; Motor Feed Pump Minimum Flow Valve Not Full Open; dated June 22, 2009
CR 09-60851; Control Room C61 Printers Not Functioning; dated June 22, 2009
CR 09-60857; Steam Jet Air Ejector Relief Lifted After Reactor Scram; dated June 22, 2009
CR 09-60855; MSR High Level Trip Signal Caused Turbine Trip & RX Scram; dated June 21, 2009
Post Scram Restart Report Perry Nuclear Power Plant; dated June 22, 2009
Operational Decision Making Issue Summary Sheet; dated June 23, 2009

Narrative Logs; dated June 21-22, 2009
Post Response Crew Self Evaluation; dated June 21, 2009
Post Scram Plant Inspection; dated June 21, 2009
DWG B-208-151 Sheet 391; Turbine Control (EHC) Emergency Trip System; Revision C

4OA7 Licensee-Identified Violations
CR 09-56298; LHRA Entry without Re-brief, dated March 30, 2009
CR 09-54406; Individual Took a Radioactive Source Off-site to Their Residence, dated February 28, 2009
CR 09-56310; Latent Design Issue With Original Seismic Qualification Documentation; dated March 30, 2009
Mode Hold Resolution Form; dated April 7, 2009
Calculation No. EA-0271; Review of Seismic Analysis of ECCS and RCIC Waterleg Pumps; dated April 7, 2009

LIST OF ACRONYMS USED

AC	alternating current
ALARA	as low as reasonably achievable
ASME	American Society of Mechanical Engineers
CAP	corrective action program
CFR	*Code of Federal Regulations*
CR	condition report
CST	condensate storage tank
ECC	emergency closed cooling
ECP	engineering change package
EDG	emergency diesel generator
ESW	emergency service water
FENOC	FirstEnergy Nuclear Operating Company
FHB	fuel handling building
HPCS	high pressure core spray
IMC	Inspection Manual Chapter
IOI	Integrated Operating Instruction
IP	Inspection Procedure
IR	Inspection Report
NOBP	Normal Operating Business Practice
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPCS	low pressure core spray
NCV	non-cited violation
NOP	Nuclear Operating Procedure
NRC	Nuclear Regulatory Commission
ODMI	Operational Decision Making Instruction
ORM	Operations Requirement Manual
OSP	Outage Safety Plan
PAF	Primary Access Facility
PAP	Perry Administrative Procedure
PI	performance indicator
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RFO	refueling outage
RHR	residual heat removal
RPV	reactor pressure vessel
SDC	shutdown cooling
SDP	Significance Determination Process
SOI	System Operating Instruction
SSC	structure, system, and component
STAR	Stop Think Act Review
SVI	Surveillance Instruction
TS	Technical Specification
TSO	transmission system operator
UFSAR	Updated Final Safety Analysis Report
UHS	ultimate heat sink
USAR	Updated Safety Analysis Report
WO	work order