UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
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May 13, 2016

Mr. David Hamilton
Site Vice President
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Perry Nuclear Power Plant
P.O. Box 97, 10 Center Road, A–PY–290
Perry, OH 44081–0097

SUBJECT: PERRY NUCLEAR POWER PLANT—NRC INTEGRATED INSPECTION REPORT 05000440/2016001

Dear Mr. Hamilton:

On March 31, 2016, the U.S. Nuclear Regulatory Commission (NRC) completed a baseline inspection at your Perry Nuclear Power Plant. On April 8, 2016, the NRC inspectors discussed this inspection with you and members of your staff. The inspectors documented the results of this inspection in the enclosed inspection report.

The NRC inspectors documented three findings of very low safety significance (Green) in this report. These findings involved violations of NRC requirements. The NRC is treating these violations as non-cited violations (NCVs), consistent with Section 2.3.2.a of the Enforcement Policy.

If you contest the violations or the significance of the NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555–0001, with copies to: (1) Regional Administrator, Region III; (2) Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001; and (3) the NRC Resident Inspector at the Perry Nuclear Power Plant.

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

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

Sincerely,

/RA/

Billy Dickson, Chief
Branch 5
Division of Reactor Projects

Docket No. 50–440
License No. NPF–58

Enclosure:
IR 05000440/2016001

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–440
License No:	NPF–58
Report No:	05000440/2016001
Licensee:	FirstEnergy Nuclear Operating Company (FENOC)
Facility:	Perry Nuclear Power Plant
Location:	North Perry, Ohio
Dates:	January 1 through March 31, 2016
Inspectors:	M. Marshfield, Senior Resident Inspector J. Nance, Resident Inspector
Approved by:	B. Dickson, Chief Branch 5 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

Inspection Report (IR) 05000440/2016001, 01/01/2016 – 03/31/2016, Perry Nuclear Power Plant; Integrated Baseline Inspection.

This report covers a 3-month period of inspection by resident inspectors and announced baseline inspections by regional inspectors. Three findings were identified. The significance of inspection findings is indicated by their color (i.e., greater than Green, or Green, White, Yellow, Red) and determined using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process," dated April 29, 2015. Cross-cutting aspects are determined using IMC 0310, "Aspects Within the Cross-Cutting Areas," dated December 4, 2014. All violations of NRC requirements are dispositioned in accordance with the NRC's Enforcement Policy, dated February 4, 2015. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG–1649, "Reactor Oversight Process," dated February 2014.

NRC-Identified and Self-Revealed Violations

Cornerstone: Initiating Events

Green. A finding of very low safety significance and an associated non-cited violation (NCV) of Technical Specification (TS) 5.4.1., "Procedures," was self-revealed on January 24, 2016, when an unplanned automatic reactor protection system (RPS) actuation occurred as a result of the licensee's failure to correctly implement the steps outlined in procedure SOI–C34, "Feedwater Control System," Section 4.2.12.c to balance inservice flow controller outputs. Specifically, while in the process of reducing power to allow for a drywell entry to determine the location of an unidentified leak into the drywell floor drain sump, the operators failed to control reactor pressure vessel water level during shifting of feedwater pumps from a turbine-driven reactor feed pump to the motor-driven reactor feed pump, resulting in a RPS actuation initiated on reactor vessel water Level 8, shutting down the reactor. Following the reactor scram, the licensee took immediate actions to restore and maintain RPV water level in accordance with procedure ONI–C71–1, "Reactor Scram," Revision 20. The issue was entered into the licensee's corrective action program as CR 2016–01063.

The licensee's failure to properly implement the steps in the procedure was a performance deficiency that was determined to be more than minor and thus a finding, because it was associated with the Initiating Events cornerstone attribute of human performance and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was determined to be of very low safety significance because it did not result in the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition. This finding has a cross-cutting aspect in the area of human performance, resources, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are available and adequate to support nuclear safety. Specifically, the licensee failed to provide adequate, procedural guidance on when to conduct the feedwater pump shift [IMC 0310, H.1]. (Section 1R20.1 b.(1))

Green. A finding of very low safety significance and an associated NCV of 10 CFR Part 50, Appendix B, Criterion IX, "Control of Special Processes," was self-revealed on January 24, 2016, for the licensee's failure to control welding and inspection activities during the replacement of the reactor recirculation loop 'A' pump discharge valve vent line during the 2015 refueling outage. When identified as the source of reactor boundary leakage in January 2016, the licensee determined that the weld did not meet the requirements on the design drawing and that the quality control (QC) inspection should have identified the non-conforming weld. The issue was entered into the licensee's corrective action program as CR 2016–01071. Corrective actions included installation of an alternative pipe and cap to replace the failed vent line appendage, plugging and capping of the reactor recirculation loop 'A' flow control valve vent line appendage and performed a weld build up on the reactor recirculation loop 'B' flow control valve vent appendage line.

The inspectors determined that the licensee's failure to control welding and inspection activities was a performance deficiency that was determined to be more than minor and thus a finding, because it was associated with the Initiating Events cornerstone attribute of human performance and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The finding was determined to be of very low safety significance because it was determined that after a reasonable assessment of degradation, the leak would not have exceeded the reactor coolant system leak rate for a small-break loss of coolant accident (LOCA) and the leak would not have affected other systems used to mitigate a LOCA (e.g., an interfacing system LOCA). This finding has a cross-cutting aspect in the area of human performance, resources, because the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to support nuclear safety. Specifically, the licensee failed to provide additional precautions, controls, and oversight for the personnel performing the welding activities, inspection activities, and supervisory activities, such that the welder, QC inspector, and supervisor were able to complete a weld that met the requirements of the design drawing and to perform an adequate inspection of the weld to determine that it met the acceptance criteria established by the design drawing [IMC 0310, H.1]. (Section 1R20.1 b.(2))

Cornerstone: Mitigating Systems

Green. A finding of very low safety significance and an associated NCV of TS 5.4.1, "Procedures," was self-revealed on January 24, 2016, when a loss of safety system function occurred as a result of the operators failing to take steps to prevent all operable average power range monitors (APRMs) from becoming out of specification in the non-conservative direction after a recirculation pump shift to slow speed. Specifically, while in the process of reducing power to allow for a drywell entry at low power, the recirculation pumps were shifted and all operable APRMs went out of specification low, which is the non-conservative direction. The operators immediately declared the APRMs inoperable and took actions to restore the operability of at least one APRM in each channel. The issue was entered into the licensee's CAP as CR 2016–01058.

The licensee's failure to take action to prevent all operable APRMs from going out of calibration low, despite understanding the cause, was determined to be more than minor and thus a finding, because it was associated with the Mitigating Systems cornerstone attribute of human performance and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The finding was determined to be of very low safety significance because it did not result in the loss of reactivity control systems beyond a single trip signal function and did not result in a mismanagement of reactivity by the operators. This finding has a cross-cutting aspect in the area of human performance, avoid complacency, for knowing that the APRMs would go out of calibration because of the pump shift but without regard for the inherent risk while expecting the successful outcome that at least one would stay in calibration without any consideration of potential actions that could have been taken to prevent the loss of safety function and reportable condition [IMC 0310, H.12]. (Section 1R20.1 b.(3))

REPORT DETAILS

Summary of Plant Status

The plant began the inspection period at 100 percent power. On January 23, 2016, at 9:00 p.m., the plant began reducing power to 8 percent power to allow a drywell entry to determine the source of elevated reactor coolant system (RCS) unidentified leakage. At 9:22 p.m., the plant determined that unidentified leakage in the drywell had increased by more than two gallons per minute (gpm) within a 24 hour period to 2.85 gpm which ultimately required a full reactor shutdown in accordance with plant Technical Specifications (TSs). At 10:07 a.m. on January 24, 2016, with the reactor at 8 percent power, an unplanned automatic reactor scram occurred due to a reactor pressure vessel (RPV) Level 8 initiation signal while shifting feedwater pumps. Plant restoration to power operations following the completion of repairs commenced on January 31, 2016, and the reactor reached criticality on the same day. The plant synchronized to the grid on February 1, 2016, and reached 100 percent power on February 3, 2016. On February 8, 2016, two safety-relief valves (SRVs) opened on what was later determined to be a spurious signal from the 'B' RPV pressure and level reference leg. The operators entered the off normal instruction for unplanned open SRVs and began to lower power to 96 percent so that the SRVs could be closed in accordance with the procedure. The SRVs remained open during these actions and caused the suppression pool average temperature to rise to 95 degrees at which point the licensed operators, following plant procedures, inserted a manual scram. The SRVs then closed as pressure decreased to below setpoints and were electrically secured until troubleshooting could be completed. Plant restoration to power operations following venting of the reference leg and associated components on February 12, 2016, and the reactor reached criticality on the same day. The plant synchronized to the grid on February 13, 2016, and reached 100 percent power on February 17, 2016. With the exception of minor reduction in power to support routine surveillances and deeper down powers for rod pattern adjustments following plant start-up, the plant remained at full power for the remainder of the quarter.

1. REACTOR SAFETY

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

1R01 Adverse Weather Protection (71111.01)

.1 Readiness for Impending Adverse Weather Condition—High Wind Conditions

a. Inspection Scope

Since high winds were forecast in the vicinity of the facility for February 29, 2016, the inspectors reviewed the licensee's overall preparations/protection for the expected weather conditions. On February 29, 2016, the inspectors also walked down the main transformer yard, the Unit 2 startup transformer area, and other areas inside the protected area, including the independent spent fuel storage installation, emergency service water structure and other safety-related equipment. These areas were walked down, in addition to the licensee's emergency alternating current power systems, because their safety-related functions could be affected or required because of high winds, tornado-generated missiles or the loss of offsite power. The inspectors evaluated the licensee staff's preparations against the site's procedures and determined that the staff's actions were adequate. During the inspection, the inspectors focused on plant-specific design features and the licensee's procedures used to respond to specified adverse weather conditions. The inspectors also toured the plant grounds to look for any loose debris that could become missiles during a tornado. The inspectors evaluated operator staffing and accessibility of controls and indications for those systems required to control the plant. Additionally, the inspectors reviewed the Updated Safety Analysis Report (USAR) and performance requirements for systems selected for inspection, and verified that operator actions were appropriate as specified by plant specific procedures. The inspectors also reviewed a sample of corrective action program (CAP) items to verify that the licensee identified adverse weather issues at an appropriate threshold and dispositioned them through the CAP in accordance with station corrective action procedures. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one readiness for impending adverse weather condition sample as defined in IP 71111.01–05.

b. Findings

No findings were identified.

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- emergency closed cooling 'B';
- annulus exhaust gas treatment 'A'; and
- residual heat removal 'A'.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety cornerstones at the time they were inspected. The inspectors attempted to identify any discrepancies that could impact the function of the system and, therefore, potentially increase risk. The inspectors reviewed applicable operating procedures, system diagrams, USAR, TS requirements, outstanding work orders (WOs), condition reports, and the impact of ongoing work activities on redundant trains of equipment in order to identify conditions that could have rendered the systems incapable of performing their intended functions. The inspectors 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 an appropriate significance characterization. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

On March 16 and 31, 2016, the inspectors performed a complete system alignment inspection of the standby liquid control system to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding WOs was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved. Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

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

- fire zones 1CC–3c, Unit 1 – Division 1 4160V and 480V switchgear room control complex 620' elevation and 1DG–1c, Unit 1 – Division 1 diesel generator building 620' and 646' elevations;
- fire zones 0FH–1, fuel handling building 574' elevation and 0FH–2a and 2b, fuel handling building 599' elevation;
- fire zone 1AB–1g, auxiliary building 574' elevation;
- fire zone 1CC–6. control complex 679' elevation; and
- fire zone 0IB–3; intermediate building 620' elevation.

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

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

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

a. Inspection Scope

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

1R11 Licensed Operator Requalification Program and Licensed Operator Performance (71111.11)

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

a. Inspection Scope

On February 8, 2016, the inspectors observed a crew of licensed operators in the plant's simulator during licensed operator requalification training. The inspectors verified that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and that training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

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

This inspection constituted one quarterly licensed operator requalification program simulator sample as defined in IP 71111.11–05.

b. Findings

No findings were identified.

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

a. Inspection Scope

On January 31, 2016, the inspectors observed the reactor plant startup from a forced outage in the control room. This was an activity that required heightened awareness or was related to increased risk. The inspectors evaluated the following areas:

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

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

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

a. Inspection Scope

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

- reactor water cleanup system;
- reactor recirculation system; and
- Unit 1 vital battery chargers.

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

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

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

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

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors reviewed the licensee's evaluation and management of plant risk for the conditions or 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:

- water intrusion into control room heating ventilation and air-conditioning system;
- reactor recirculation pump failure to shift to fast speed;
- Division 1 safety-related bus EH11, breaker EH1103 fuse failure;
- average power range monitor ‘A’ multiplexer control card failure;
- preferred source breaker EH1114 for bus EH11; and
- troubleshooting of direct current Division ‘B’ ground condition.

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. Documents reviewed during this inspection are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- main supply breaker EF2A03 from bus EH21 arc chute crack;
- safety related breaker overhauls overdue;
- operation following pressure transient on reference leg ‘B’;
- unplanned outage shutdown safety risk; and
- at-power operation with tailpipe leakage from one SRV.

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

This operability inspection constituted five samples as defined in IP 71111.15–05.

b. Findings

No findings were identified.

1R18 Plant Modifications (71111.18)

a. Inspection Scope

The inspectors reviewed a permanent modification which changed the grounding scheme for the current transformers on the reactor recirculation pump 5A breaker as a correction for the pump's failure to start in fast speed and based on comparison by the licensee to the other pump's breaker wiring.

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

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

b. Findings

No findings were identified.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- reactor mode switch repair PMT;
- generator stator cooling water pump ‘B’ replacement PMT;
- reactor recirculation pump ‘A’ PMT;
- average power range monitor (APRM) ‘A’ multiplexer control card replacement PMT;
- Division 1 safety-related bus EH11 preferred source breaker EH1114 replacement PMT; and
- Division 1 emergency diesel generator outage PMT.

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

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

b. Findings

No findings were identified.

1R20 Outage Activities (71111.20)

.1 Other Outage Activities – 1FOAC1

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled outage, 1FOAC1, that began on January 24, 2016, and continued through February 1, 2016. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, startup and heatup activities, and identification and resolution of problems associated with the outage. The outage was a forced shutdown caused by leakage in excess of the TS allowance for unknown leakage into the drywell, specifically an increase of more than two gallons per minute in less than a 24-hour period. The leakage site was identified upon entry to the drywell, following an unplanned scram, on a vent piping weld on the "A" recirculation loop. The reactor was cooled down and the weld repaired prior to restart on January 31, 2016. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one other outage sample as defined in IP 71111.20–05.

b. Findings

(1) Failure to Properly Implement System Operating Instructions to Maintain Control of Reactor Pressure Vessel Level

Introduction: A finding of very low safety significance (Green) and an associated NCV of TS 5.4.1, "Procedures," was self-revealed on January 24, 2016, when an unplanned automatic Reactor Protection System (RPS) actuation occurred. Specifically, the operators failed to correctly implement the steps outlined in procedure SOI–C34, "Feedwater Control System," Section 4.2.12.c, to balance inservice flow controller outputs. As a result of the operators' failure to control RPV water level while shifting feedwater pumps from a turbine-driven reactor feed pump (RFPT) to the motor-driven reactor feed pump (MFP), a RPS actuation initiated on RPV water Level 8, shutting down the reactor.

Description: On January 24, 2016, the licensee had reduced power from full power to approximately eight percent power to support a drywell entry to identify a RCS leak. The licensee had been monitoring this leak and the leak had increased from 0.7 gallons per minute (gpm) on January 19, 2016, to 2.85 gpm on January 24, 2016. At approximately 10:03 p.m., the licensee began shifting feedwater flow from the RFPT to the MFP. Both feed pumps were in manual control due to being at eight percent power. Procedure SOI–C34, "Feedwater Control System," Section 4.2.12.c, requires the licensee to balance the inservice flow controller outputs while shifting feedwater pumps. As the licensee began feeding the RPV with the MFP, an anticipated slow rise in RPV level was observed. The licensee reduced the RFPT controller output by one percent. The one-percent controller output change resulted in a faster than anticipated lowering trend in the RPV water level. The licensee began increasing MFP flow to the RPV in response to the lowering water level. At approximately 10:05 p.m., the licensee was successful in stopping the lowering vessel trend; however, the MFP was now feeding the RPV at a rate that resulted in an increasing water level trend. As the RPV water level neared its original level, the licensee began reducing the RFPT controller output in an attempt to stabilize RPV water level. Unbeknownst to the operator, however, the original one-percent reduction in RFPT controller output signal had secured all flow to the RPV from the RFPT. The licensed operators were unaware that the RFPT was no longer feeding the RPV because they were not monitoring individual feed pump flow indications. The licensed operators were focused on total flow to the RPV. The result was that the licensee's attempts to stabilize RPV water level by reducing feedwater flow from the RFPT was having no impact on the feed flow to the RPV and therefore no impact on the increasing RPV water level. RPV water level continued to rise to Level 8 where the RPS actuated, shutting down the reactor.

Analysis: The inspectors determined that the licensee's failure to correctly implement the steps outlined in procedure SOI–C34, "Feedwater Control System" for transferring feedwater flow from a RFPT to the MFP was a performance deficiency warranting further review. Using the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, Issue Screening," dated September 7, 2012, the inspectors determined that the performance deficiency was more than minor; and thus a finding because it was associated with the Initiating Events cornerstone attribute of human performance and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined that the finding could be evaluated using the significance determination process in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 1, dated June 19, 2012. The inspectors reviewed the Initiating Events screening questions in Exhibit 1 and answered "no" to the question, "Did the finding cause a reactor trip AND the loss of mitigation equipment relied upon to transition the plant from the onset of the trip to a stable shutdown condition (e.g., loss of condenser, loss of feedwater)?" Therefore, the finding screened as very low safety significance (Green). The MFP did trip as expected on a Level 8 RPS actuation but was recovered by the operators as soon as the level dropped below the point where the MFP could be restarted.

This finding has a cross-cutting aspect in the area of human performance, resources, because the licensee failed to ensure that personnel, equipment, procedures, and other resources are available and adequate to support nuclear safety. Specifically, the licensee failed to provide adequate, procedural guidance on when to conduct the feedwater pump shift during the shutdown process [IMC 0310, H.1]

Enforcement: Technical Specification 5.4.1, "Procedures," requires in part, that written procedures and instructions be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide (RG) 1.33, Revision 2, Appendix A, dated February 1978. RG 1.33, Revision 2, Appendix A, Section 4, states, in part, that instructions for energizing, filling, venting, draining, startup, shutdown, and changing modes of operation should be prepared, as appropriate, for the feedwater system (feedwater pumps to the reactor vessel). Contrary to this requirement, on January 24, 2016, while shifting feedwater pumps from a turbine-driven reactor feed pump to the motor-driven feed pump, the licensee failed to correctly implement the step outlined in the procedure and caused an automatic RPS actuation, shutting down the reactor. Following the reactor scram, the licensee took immediate actions to restore and maintain RPV water level in accordance with procedure ONI–C71–1, "Reactor Scram," Revision 20. The licensee entered the issue into its CAP as CR 2016–01063. Because this violation was of very low safety significance and was entered into the licensee's CAP, this violation is being treated as a NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy.

(NCV 05000440/2016001–01, Failure to Properly Implement System Operating Instructions to Maintain Control of Reactor Pressure Vessel Level)

(2) Failure to Control Welding and Inspection Activities to Maintain Reactor Coolant System Integrity

Introduction: A finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion IX, "Control of Special Processes," was self-revealed on January 24, 2016, for failing to control welding and inspection activities during the replacement of the reactor recirculation loop 'A' pump discharge valve vent line during the 2015 refueling outage. When identified as the source of reactor boundary leakage in January 2016, the licensee determined that the weld did not meet the requirements on the design drawing and that the post-weld quality control inspection should have identified the non-conforming weld.

Description: On January 19, 2016, the licensee identified an increase in RCS leakage that had been stable at 0.7 gpm. As the unidentified leakage continued to increase over the next four days, the licensee began planning for a forced outage to identify where the leak was coming from and to repair the leak. On January 23, 2016, at 9:00 p.m., the licensee began shutting down the plant from one hundred percent power. At 9:22 p.m., after the licensee began reducing power by reducing flow through the reactor recirculation loops, operators saw the RCS unidentified leakage rate increase to 2.85 gpm. The increase in the RCS unidentified leak rate exceeded the TS 3.4.5.d limit of less than or equal to 2 gpm increase in unidentified leakage within the previous 24 hour period while in Mode 1, which requires the licensee to verify that the source of the unidentified leakage increase is not from service sensitive austenitic material within four hours or be shutdown in twelve hours and cooled down in thirty-six hours. As the licensee was preparing to enter the drywell at eight percent power to identify and repair the leak, the RPS actuated on a valid reactor pressure vessel high water level signal and scrammed the reactor. When the licensee entered the drywell following the reactor scram, the leak was identified as an un-isolable leak from the RCS emanating from a weld on the vent line appendage of the reactor recirculation loop 'A' discharge valve. That appendage was replaced during the refueling outage in the spring of 2015.

The licensee identified the root cause of the weld failure as a failure to effectively manage the risk associated with installing a three-quarter inch vent appendage on the reactor recirculation system, resulting in a deficient weld and recurrence of a previous similar failure of the RCS pressure boundary. The licensee's root cause report further stated:

> "No additional process requirements or barriers were added as a result of previous appendage weld failures to ensure the integrity of the reactor coolant pressure boundary was not compromised by a repeat failure.
>
> - The organization did not aggressively pursue the 5/2/1990 SIL 512 recommendations to remove or support appendages.
> - A legacy issue that was not recognized and adequately addressed.
> - The failed weld was performed as a standard weld with no additional oversight.
> - No additional precautions or controls were in the work planning process for working on the reactor coolant pressure boundary.

- Performing a recommended mock-up prior to implementing the modification did not occur.
- No heightened instruction or additional oversight of the workers.
- The weld inspection was performed by a supplemental inspector with no additional oversight.
- By procedure, NOP–WM–1001, interface with the welding engineer is at the discretion of the planner, no additional instruction is included for work on the reactor coolant boundary"

Analysis: The inspectors determined that the licensee's failure to control welding and inspection activities was a performance deficiency warranting further review. Using the guidance in IMC 0612, "Power Reactor Inspection Reports," Appendix B, Issue Screening," dated September 7, 2012, the inspectors determined that the performance deficiency was more than minor; and thus a finding because it was associated with the Initiating Events cornerstone attribute of human performance and adversely affected the cornerstone objective to limit the likelihood of events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. The inspectors determined that the finding could be evaluated using the significance determination process in accordance with IMC 0609, Appendix A, "The Significance Determination Process for Findings At-Power," Exhibit 1, dated June 19, 2012. The inspectors reviewed the Initiating Events screening questions in Exhibit 1 and determined the finding to be of very low safety significance (Green) because after a reasonable assessment of degradation the leak would not have exceeded the RCS leak rate for a small-break LOCA and the leak would not have affected other systems used to mitigate a LOCA (e.g., an interfacing system LOCA).

This finding has a cross-cutting aspect in the area of human performance, resources, because the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to support nuclear safety. Specifically, the licensee failed to provide additional precautions, controls, and oversight for the personnel performing the welding activities, inspection activities, and supervisory activities, such that the welder, QC inspector, and supervisor were able to complete a weld that met the requirements of the design drawing and to perform an adequate inspection of the weld to determine that it met the acceptance criteria established by the design drawing [IMC 0310, H.1].

Enforcement: Title 10 of the CFR, Part 50, Appendix B, Criterion IX, "Control of Special Processes," requires that measures shall be established to assure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements." Contrary to this requirement, on or about April 15, 2015, the licensee failed to control welding and inspection activities during the replacement of the vent valve appendage on the reactor recirculation loop 'A' discharge valve. Specifically, the licensee failed to assure that the welding and inspection activities were completed per the design drawing. More specifically, the fillet weld legs of the shop and field welds were butted together in some instances and had a space between in others with no apparent weld buildup on the pipe as directed by the drawing. The licensee entered the issue into its CAP as CR 2016–01071. Correction actions included installation of an alternative pipe and cap to replace the failed vent line appendage, plugging and capping of the reactor recirculation loop 'A' flow control valve vent line appendage and performed a weld build up on the reactor recirculation loop ‘B’ flow control valve vent appendage line. Because this violation was of very low safety significance and was entered into the licensee’s CAP, the violation is being treated as a NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000440/2016001–02, Failure to Control Welding and Inspection Activities to Maintain Reactor Coolant System Integrity)**

(3) <u>Failure to Take Actions to Prevent a Loss of Safety Function during Reactor Recirculation Pump Downshift</u>

<u>Introduction</u>: A finding of very low safety significance (Green) and an associated NCV of TS 5.4.1, “Procedures,” was self-revealed on January 24, 2016, when a loss of safety system function occurred as a result of the operators failing to take steps to prevent all operable APRMs from becoming out of specification in the non-conservative direction after a recirculation pump shift to slow speed. Specifically, while in the process of reducing power to allow for a drywell entry at low power the recirculation pumps were shifted to slow speed and all seven of the inservice APRMs went out of specification low which was the non-conservative direction. The operators immediately declared all inservice APRMs inoperable and took actions to restore the operability of at least one APRM in each channel.

<u>Description</u>: On January 24, 2016, while shutting the reactor down for a drywell entry to locate a leak inside the drywell, the licensee conducted a downshift of reactor recirculation pumps. Prior to the downshift, the operators took no action to calibrate APRMs so that the post-shift condition of the APRMs would remain operable. The downshift process is an understood process by the operators, which causes the reference calculation for calibration of the APRMs to shift to a different source. This is known to cause the APRMs to read a lower value than the reference. The resulting shift of the pumps with no operator actions caused all seven of the inservice APRMs to read greater than 2 percent lower than the calculated reference. As a result, all inservice APRMs were declared inoperable and the operator made appropriate reports to the NRC for the loss of safety function. The operators took immediate actions to restore at least one APRM in each of the two channels of the RPS.

<u>Analysis</u>: The inspectors determined that the operator’s failure to take action to prevent all operable APRMs from going out of specification low, despite understanding the cause, was a performance deficiency warranting further review. Using the guidance in IMC 0612, “Power Reactor IRs,” Appendix B, “Issue Screening,” dated September 7, 2012, the inspectors determined that the performance deficiency was more than minor; and thus, a finding because it was associated with the human performance attribute of the Mitigating Systems cornerstone and adversely affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The inspectors determined that the finding could be evaluated using the significance determination process in accordance with IMC 0609, Appendix A, “The Significance Determination Process for Findings At-Power,” Exhibit 2, dated June 19, 2012. The inspectors reviewed the Mitigating Systems screening questions in Exhibit 2 for Reactivity Control Systems and determined the finding to be of very low safety significance because it did not result in the loss of reactivity control systems beyond a single trip signal function and did not result in a mismanagement of reactivity by the operators. (Green).

The finding has a cross-cutting aspect in the area of human performance, avoid complacency, for the licensee's failure to take corrective action while knowing that downshifting the recirculation pumps would cause the calculation process for total power to shift to a different reference source impacting the operability of the APRMs. Specifically, despite operational knowledge that the change in reference for the total power calculation would occur as a result of the pump downshift to slow speed and likely cause APRMs to read low, no preparatory actions were taken or even considered by operations personnel in the control room prior to the pump shift [IMC 0310, H.12].

Enforcement: Technical Specification 5.4.1, "Procedures," requires in part, that written procedures and instructions be established, implemented, and maintained covering the applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, dated February 1978. Regulatory Guide 1.33, Revision 2, Appendix A, Section 4, states, in part, that instruction for energizing, filling, venting, draining, startup, shutdown, and changing modes of operation should be prepared as appropriate for shutdown cooling and reactor vessel head spray system. Further, procedure NOP–OP–1002, "Conduct of Operations," requires operators to control plant evolutions in accordance with approved and up-to-date procedures, clearances, and other documents as appropriate to maintain proper configuration control and reduce the potential for operational events. Contrary to these requirements, on January 24, 2016, while performing a reactor system shutdown during the downshifting of reactor recirculation pumps, the licensee failed to take action to prevent a loss of safety function when all inservice APRMs were rendered inoperable by the operators' failure to take action prior to the shift.

Following the downshift of the reactor recirculation pumps to slow speed, the licensee took immediate actions to restore at least one APRM in each channel to an operable status and exited the condition. Further actions were taken to strengthen procedural guidance so that future evolutions will not result in a similar occurrence. The issue was entered into the licensee's CAP as CR 2016–01058. Because this violation was of very low safety significance and it was entered into the licensee's CAP, this violation is being treated as an NCV, consistent with Section 2.3.2.a of the NRC Enforcement Policy. **(NCV 05000440/2016001–03, Failure to Take Actions to Prevent a Loss of Safety Function during Reactor Recirculation Pump Downshift)**

.2 Other Outage Activities – 1FOAC2

a. Inspection Scope

The inspectors evaluated outage activities for an unscheduled outage, 1FOAC2, that began on February 8, 2016, and continued through February 13, 2016. The inspectors reviewed activities to ensure that the licensee considered risk in developing, planning, and implementing the outage schedule.

The inspectors observed or reviewed the reactor shutdown and cooldown, outage equipment configuration and risk management, electrical lineups, selected clearances, control and monitoring of decay heat removal, control of containment activities, personnel fatigue management, startup and heatup activities, and identification and resolution of problems associated with the outage. The outage was caused when two safety relief valves inadvertently opened and caused a heatup of the suppression pool before operator actions could be completed to reclose the valves. The operators manually inserted a scram based on temperature of the suppression pool in accordance with their procedures. During the forced outage, additional events occurred which led the NRC to dispatch a special inspection team to evaluate the actions of the licensee in response to the scram and during the restart. The results of the special inspection team's activities will be documented in a separate report. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one other outage sample as defined in IP 71111.20–05.

b. Findings

No findings were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- SVI–C71–T0427, "RX Mode Switch Refuel Mode Channel Functional" (Routine);
- SVI–P47–T2001A, "Control Complex Chilled Water 'A' Pump and Valve Operability Test" (Inservice Testing);
- Leakage detection activities conducted to determine a source of leakage into the drywell (RCS Leakage); and
- SVI–C51–T0030–G, "APRM 'G' Channel Calibration for 1C51–K605G" (Routine)

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

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

Mechanical Engineers code, and reference values were consistent with the system design basis;

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

Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

1EP6 Drill Evaluation (71114.06)

Training Observation

a. Inspection Scope

The inspector observed a simulator training evolution for licensed operators on February 8, 2016, which required emergency plan implementation by a licensee operations crew. This evolution was planned to be evaluated and included in performance indicator (PI) data regarding drill and exercise performance. The inspectors observed event classification and notification activities performed by the crew. The inspectors also attended the post-evolution critique for the scenario. The focus of the inspectors' activities was to note any weaknesses and deficiencies in the crew's performance and ensure that the licensee evaluators noted the same issues and entered them into the CAP. As part of the inspection, the inspectors reviewed the scenario package and other documents listed in the Attachment to this report.

This inspection of the licensee's training evolution with emergency preparedness drill aspects constituted one sample as defined in IP 71114.06–06.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

Cornerstones: Initiating Events, Mitigating Systems, Barrier Integrity, Emergency Preparedness, Public Radiation Safety, Occupational Radiation Safety, and Security

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 PI for the period from the first quarter 2015 through the fourth quarter 2015. To determine the accuracy of the PI data reported, definitions and guidance contained in Nuclear Energy Institute (NEI) 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, were used. The inspectors reviewed licensee's operator narrative logs, issue reports, event reports and NRC integrated inspection reports (IRs) to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned scrams per 7000 critical hours (IE01) sample as defined in IP 71151–05.

b. Findings

No findings 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 the first quarter 2015 through the fourth quarter 2015. To determine the accuracy of the PI data reported, definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, was used. The inspectors reviewed licensee's operator narrative logs, issue reports, event reports and NRC integrated IRs to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned scrams with complications (IE04) sample as defined in IP 71151–05.

b. Findings

No findings were identified.

.3 Unplanned Power Changes per 7000 Critical Hours

a. Inspection Scope

The inspectors sampled licensee submittals for the Unplanned Power Changes per 7000 Critical Hours PI for the period from the first quarter 2015 through the fourth quarter 2015. To determine the accuracy of the PI data reported, definitions and guidance contained in NEI Document 99–02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013, was used. The inspectors reviewed licensee's operator narrative logs, issue reports, maintenance rule records, event reports and NRC integrated IRs to validate the accuracy of the submittals. The inspectors also reviewed the licensee's issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none were identified. Documents reviewed are listed in the Attachment to this report.

This inspection constituted one unplanned power changes per 7000 critical hours (IE03) sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.3 Semi-Annual Trend Review

a. Inspection Scope

The inspectors performed a review of the licensee's CAP and associated documents to identify trends that could indicate the existence of a more significant safety issue. The inspectors' review was focused on repetitive equipment issues, but also considered the results of daily inspector CAP item screening discussed in Section 4OA2.2 above, licensee trending efforts, and licensee human performance results. The inspectors' review nominally considered the 6-month period of July 1, 2015, through December 31, 2015, although some examples expanded beyond those dates where the scope of the trend warranted.

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

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

b. Findings

No findings were identified.

.4 Annual Follow-up of Selected Issues: Reviewed Licensee Corrective Actions for Inadequate Diesel Generator Ventilation System Operating Procedure Identified by NRC NCV

a. Inspection Scope

During a review of items entered in the licensee's CAP, the inspectors recognized two corrective action items (CRs 2015–11597 and 2016–00503) documenting the licensee's corrective actions in response to a NCV identified in NRC integrated IR 05000440/2015003, dated November 10, 2015. The NCV documented the licensee's failure to establish and maintain an adequate procedure for the operation of the diesel generator building ventilation system to ensure that the diesel room temperature would remain below limits during testing.

The inspectors reviewed the corrective actions taken in CR 2015–11597. The licensee took action to revise procedure SOI–M43, "Diesel Generator Building Ventilation System," Revision 13, with an effective date of September 22, 2015, to ensure that the ventilation system would be operated as described in the USAR. During their reviews of CR 2015–11597 and CR 2016–00503, the inspectors made the following observations.

- CR 2015–11597, "Potential NRC Violation concerning operation of the DG ventilation fans," dated September 2, 2015, was processed as a Category-AF (adverse fix). Procedure NOP–LP–2001, "Corrective Action Program," Revision 37, states, in part, in Attachment 2, "Condition Report Evaluation Methods," that "Fix – Evaluation Code 'F' …is not sufficient for process, program, or equipment issues that result in: NRC cited/non-cited violation".

- CR 2016–00503, "NRC NCV – Inadequate procedure for Operations of DG Building Ventilation System," dated January 13, 2016, was initiated and cancelled on the same day. Procedure NOP–LP–2001 states, in 4.3.5, in part, that "CRs shall be written to document receipt of NRC Findings or Cited or Non-Cited Violations in accordance with procedure NOBP–LP–4014 to specifically address the issue(s) as stated in the wording received from the NRC." Procedure NOBP–LP–4014, "Managing Regulatory Interface," Revision 6, states, in 4.1.7, in part, that "Issues may already be documented in the corrective action program, but the NRC characterization may have changed or otherwise be different…" It further states, in part that "Condition reports initiated for any findings, violations, or non-cited violations associated with cross-cutting aspects in the areas of Human Performance, PI&R, and SCWE should be recommended for a causal evaluation at a minimum, since they are associated with violations of regulatory requirements or NRC findings of more than minor significance. Performance of a casual evaluation will also ensure that a latent organizational weakness evaluation is performed, since cross-cutting aspects reflect breakdowns across organizational boundaries." The licensee did not conduct a causal evaluation to review the assigned cross-cutting aspect as described in licensee procedures.

The inspectors concluded that these were minor findings as there was only one example of the licensee's failure to follow a quality procedure and there is no regulatory requirement to write a condition report to address individual cross-cutting aspects assigned to a NCV.

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

b. Findings

No findings were identified.

Exit Meeting Summary

On April 8, 2016, the inspectors presented the inspection results to Mr. D. Hamilton, and other members of the licensee staff. The licensee acknowledged the issues presented. The inspectors confirmed that none of the potential report input discussed was considered proprietary. One piece of proprietary material reviewed during the quarter was returned to the licensee.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

D. Hamilton, Site Vice-President
F. Payne, Site Operations Director
T. Brown, Performance Improvement Director
D. Reeves, Site Engineering Director

U.S. Nuclear Regulatory Commission

B. Dickson, Chief, Reactor Projects Branch 5

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened and Closed

05000440/2016001–01	NCV	Failure to Properly Implement System Operating Instructions to Maintain Control of Reactor Pressure Vessel Level (Section 1R20.1 b.(1))
05000440/2016001–02	NCV	Failure to Control Welding and Inspection Activities to Maintain Reactor Coolant System Integrity (Section 1R20.1 b.(2))
05000440/2016001–03	NCV	Failure to Take Actions to Prevent a Loss of Safety Function during Reactor Recirculation Pump Downshift (Section 1R20.1 b.(3))

Discussed

None

LIST OF DOCUMENTS REVIEWED

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

1R01 Adverse Weather Protection
- eSOMS Plant Narrative Logs; Dated February 29 and March 1, 2016
- ONI–ZZZ–1; Tornado or High Winds; Revision 26
- PDB–R0003; Flex Specifications; Revision 0

1R04 Equipment Alignment
- VLI–P42; Emergency Closed Cooling System; Revision 20
- VLI–M28; Emergency Closed Cooling Pump Area Cooling System; Revision 5
- VLI–C41; Standby Liquid Control System; Revision 8
- SOI–M15; Annulus Exhaust Gas Treatment System; Revision 11
- VLI–M15; Annulus Exhaust Gas Treatment System (Unit 1); Revision 4
- ELI–R22; 15kV and 5kV Metal Clad Switchgear; Revision 9
- Dwg 912–0605–00000; Reactor Building Annulus Exhaust Gas Treatment; Revision X
- SOI–E12; Residual Heat Removal System; Revision 66
- VLI–E12; Residual Heat Removal System; Revision 14
- Dwg 302–0641–00000; Residual Heat Removal System; Revision GGG

1R05 Fire Protection
- FPI–0CC; Control Complex; Revision 9
- FPI–1DG; Diesel Generator Building; Revision 7
- FPI–0FH; Fuel Handling Building; Revision 4
- CR 2016–03722; NRC Identified Housekeeping Concerns on the Auxiliary and Intermittent Building 574' Elevations; Dated March 21, 2016
- FPI–0IB; Intermediate Building; Revision 8
- FPI–1AB; Auxiliary Building Unit 1; Revision 3

1R06 Flooding
- PRA–PY1–FP–R0b; Perry Nuclear Power Plant Probabilistic Risk Assessment IF–001 – Internal Flooding Notebook; Dated December 20, 2012
- PRA–PY1–FP–R0b; Perry Nuclear Power Plant Probabilistic Risk Assessment QU-001 – Quantification Notebook; Dated December 20, 2012
- Calculation No. JL–127; Flooding Analysis at 620' – 6" Elevation of the Control Complex; Dated August 31, 2006

1R11 Licensed Operator Requalification Program
- OTLC-3058201606 PY–SGC2–Cycle 6 2016 Evaluated Scenario C2; Revision 0

1R12 Maintenance Effectiveness
- Perry Nuclear Power Plant, Plant Health Report 2015–02 – G33 – Reactor Water Clean Up
- CR 2015-09940; Reactor Water Cleanup (G33) System Health Report Rated Red for the 1st Half of 2015; Dated July 23, 2015
- CR 2015–07039; Reactor Water Cleanup Pump Seal Leaks – Recurring Issue; Dated May 15, 2015

- ECP 15–0280; Replace Reactor Water Cleanup Pumps with Canned Rotor Pumps; Draft undated
- ECP 05–0144; Rebuild/Replace the Non-Safety Manual Remote Operators for 1G33F0005A/B, F0013A/B and F0043A/B; Dated September 26, 2005
- ECP 05–0180; Replace Valve 1G33F0033 with CCI Drag Type Valve; Dated March 30, 2010
- CR 2016–01071; Reactor Recirculation Loop A Pump Discharge Valve Vent Line Leakage; Dated January 24, 2016
- CR 2016–00949; Reactor Recirculation (B33) System Health Report Identified as RED during 2nd Half of 2015; Dated January 21, 2016
- SOI–B33; Reactor Recirculation System; Revision 35
- CR 2016–02358; Actuator Drain Alarm Received for FCV B; dated February 18, 2016
- CR 2016–02385; FCV B Actuator Drain Alarming during BCV B Movement; Dated February 18, 2016
- CR 2011–88991; Minor Cracking Noted on a Number of Dells in the Unit 1 Division 3 Battery; Dated January 31, 2011
- CR 2012–12715; Additional Unit 1, Division 3 Battery Cells Found with Cracks in the Tops of the Cell Jars; Dated August 17, 2012
- CR 2013–07576; Unit 1 Division 3 Battery Deterioration; Dated May 14, 2013
- CR 2014–15114; Part 21 for C&D KDR/LCR/LCY Batteries; Dated September 30, 2014
- CR 2015–01152; Unit 1 Division 1 Normal Battery Charger Failed to Maintain Normal Output Voltage during SVI–R42T5228; Dated January 27, 2015
- CR 2016–02060; Division 1 DC Normal Charger did not Respond as Expected; Dated February 12, 2016
- CR 2016–02441; Electrical Transient during Shift from Charger EFD–12–A to Charger EFD–1A; Dated February 20, 2016

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>
- NOP–OP–1007; Risk Management; Revision 22
- NOP–OP–1007–01; Risk Management Plan for WO 200632932, Racking Activities Associated with Breaker EH1114 E11 Preferred Source Breaker; Dated February 20, 2016
- WO 200632932; Replace Breaker EH1114 with a Refurbished Spare Breaker to Resolve the NRC 10 CFR Part 21 Notification Issue with the ABB HK Circuit Breaker Close Latch Spring Identified on CR 2015–02193; Dated February 24, 2016
- NOBP–LP–4003A; FENOC 10 CFR 50.59 User Guidelines; Revision 7
- CR 2016–00266; Water Intrusion into the Control Room HVAC & Emergency Recirculating System Train A; Dated January 8, 2016
- Drawing 208-0206-00005; Metal Clad Switchgear (15KV and 5KV) Interbus Transformer LH–1–A Supply Breaker – L1010; Revision P
- APRM ‘A’ WO 200673585 Challenge Meeting Information and Agenda; Dated February 17, 2016
- NOP–OP–1007–01; Risk Management Plan for WO 2006673585, Shutting Down the Reactor Recirculation HPUs to Prevent Recirculation Flow Control Valve Movement per SOI–B33; Dated February 17, 2016
- NOBP–OP–0007–01; IPTE Worksheet – Lock up of Both Recirculation Flow Control Valves to Allow Down Powering PS23 for APRMs A and E; Dated February 17, 2016

<u>1R15 Operability Determinations and Functionality Assessments</u>
- CR 2015–17263; ARC Chute Crack Identified on Breaker Main Supply Brkr From Bus EH21 (EF2A03) During Breaker Exercise and Service Inspection; Dated December 28, 2015
- NOP–LP–2001; Corrective Action Program; Revision 27
- NOP–LP–4003; Evaluation of Changes, Tests and Experiments; Revision 7

- NOP–OP–1010; Operational Decision-Making; Revision 6
- CR 2016–01231; ODMI Determine the Appropriate Plant Conditions to Correct Leakage Found on the Vent Valve Appendage of 1B33F0067A, Reactor Recirculation Pump ‘A’ Discharge Valve; Dated January 27, 2016
- Daily Shutdown Safety Risk Evaluations During the January 2016 Forced Outage
- CR 2016–00466; Safety Related Breaker Overhaul Past due; Dated January 12, 2016
- CR 2016–01934; ODMI Operation Following Pressure Transient on Reference Leg ‘B’; Dated February 9, 2016
- ODMI Summary Sheet; Operation Following Pressure Transient on Reference Leg B; Dated February 11, 2016
- CR 2016–02158 CR Tracking Completion of ODMI for Operation with Minor SRV 1B21F0047H Leakage; Dated February 14, 2016
- ODMI Summary Sheet; Operation with Safety Relief Valve (SRV) 1B21F0047H Leakage, Dated February 17, 2016
- CR 2016–01231; CR Tracking Completion of ODMI for Determining the Plant Conditions to Correct Leakage from Vent Valve Appendage; Dated January 27, 2016

1R18 Plant Modifications
- ECP 16–0085-000; Change Grounding Scheme for CTs on 5A Breaker (1R22S0002–006); Dated February 15, 2016

1R19 Post-Maintenance Testing
- WO 200671775; Reactor Mode Switch Simple Troubleshooting Plan; Dated January 28, 2016
- eSOMS Plant Narrative Logs; Dated January 27 – 28, 2016
- NOP–OP–1002; Conduct of Operations; Revision 11
- WO 200294684; Generator Stator Cooling Water ‘B’ Pump Replacement; Dated January 30, 2016
- WO 200632932; Replace Breaker EH1114 with a Refurbished Spare Breaker to Resolve the NRC 10 CFR Part 21 Notification Issue with the ABB HK Circuit Breaker Close Latch Spring Identified on CR 2015–02193; Dated February 24, 2016
- GEI–0135; ABB Power Circuit Breakers 5 KV Types 5HK250 and 5HK350 Maintenance; Revision 34
- eSOMS Plant Narrative Logs Dated February 23, 2016
- CR 2016–02156; Recirc Pump ‘A’ did not Transfer to Fast Speed; Dated February 14, 2016
- WO 200673305; Troubleshooting Reactor Recirculation Pump ‘A’ Incomplete Transfer to Fast Speed; Dated February 14, 2016
- Simple Troubleshooting Plan; On 02/13/2016, at 2138 Reactor Recirculation Pump ‘A’ Failed to Upshift due to Lockout; Dated February 13, 2016
- WO 200673585; APRM ‘A’/’E’ Troubleshooting and Circuit Card Replacement; Dated February 18, 2016
- SVI–R43–T1317; Diesel Generator Start and Load Division 1; Dated March 21, 2016
- SOI–R43; Division 1 and 2 Diesel Generator System; Revision 45

1R20 Outage Activities
- CR 2016–01142; Potential Impact to the A RCIRC Pump Motor; Dated January 25, 2016
- CR 2016–01141; PY–1B33F0067A Critical Path Work Stoppage; Dated January 25, 2016
- CR 2016–01060; Heater 3A Isolated on High Level Following a Reactor Recirculation Pump Downshift; Dated January 24, 2016
- CR 2016–01111; IRM E Reading 35% of Scale on Range 2 While Plant is Shutdown; Dated January 25, 2016

- CR 2016–01120; Post Event Crew Critique for Reactor Scram due to Level Transient during Feed Pump Shift; Dated January 25, 2016
- CR 2016–01055; Reactor Recirc HPU A Locked up While Opening Flow Control Valve A; Dated January 24, 2016
- CR 2016–01101; Unexpected ½ Scram when Placing Mode Switch in Refuel; Dated January 25, 2016
- CR 2016–01077; Elevated Drywell Atmosphere Radiation Monitor Particulate Channel Following 1/24/2016 Scram; Dated January 24, 2016
- CR 2016–01058; APRMs out of Calibration (Delta was >2.0% from Reactor Power) in a Non-Conservative Manner Following Reactor Recirculation Pump downshift; Dated January 24, 2016
- CR 2016–01082; First Attempt to Close 1B33F067A Torqued Out; Dated January 25, 2016
- Drawing 302–0606–00000; Nuclear Boiler System; Revision FF
- Drawing 814–0601–00102; Nuclear Boiler System, Reactor Building; Revision E
- Drawing 803–0010–00009; Instrument Loop Diagram, Nuclear Boiler System Reactor Vessel Pressure 1B21–N068B; Revision K
- Drawing 814–0605–00903; Nuclear Boiler System – Reactor Building, Reactor Vessel Level & Pressure Instrument Panel 1H22–P027; Revision A
- Drawing 814-0605–00917; Nuclear Boiler System – Reactor Building, 1H22–P027 Reactor Vessel Level & Pressure Instrument Panel 'B'; Revision A
- Drawing 814–0605–00102; Piping Isometric, Nuclear Boiler System, Reactor Building; Revision F
- Drawing D–302–607; Piping System Diagram, Nuclear Boiler System; Revision K
- CR 2016–02059; Not all Valves Closed during Expected BOP Isolation; Dated February 12, 2016
- Perry Nuclear Power Plant SRV Lifting Forced Outage Work Schedule; Dated February 10, 2016
- Perry Nuclear Power Plant SRV Lifting Forced Outage Work Schedule; Dated February 12, 2016
- ONI–B21–1; SRV Inadvertent Opening/Stuck Open; Revision 11
- Reactivity Plan – Perry Nuclear Power Plant; Evolution Specific – Startup 126 – 37–100% Revision 1, Update 0; Dated February 14, 2016
- CR 2016–02011; 1B21N0062B Pressure Spike for 10 Milliseconds; Dated February 11, 2016
- CR 2016–02031; ED1B Ground Fault Locked in; Dated February 11, 2016
- CR 2016–02061; 1D23K152A Gross Fail will not Reset; Dated February 12, 2016
- CR 2016–02051; Emergency Service Water Pump 'A' Failed to Start on Automatic Start Signal; Dated February 11, 2016
- CR 2016–02060; Div 1 DC Normal Charger did not Respond as Expected; Dated February 12, 2016
- CR 2016–02052; Unplanned Change in Shutdown Safety Risk – from Green to Yellow; Dated February 11, 2016
- Abnormal Indications on RCIS Following Power Restoration to EH11; Dated February 12, 2016
- CR 2016–02049; Division 1 Diesel Generator Ran Approximately 3 Minutes Without Emergency Service Water Cooling; Dated February 11, 2016
- Perry Nuclear Power Plant, February 2016 Forced Outage Restart Readiness Meeting Package; Dated February 10, 2016
- CR 2016–02088; EH1103 Fuse had Inconsistent Resistance Readings; Dated February 12, 2016
- CR 2016–02129; Control Rod 26–47 Experienced Position Indication Problems; Dated February 13, 2016

- CR 2016–02195; Stator Windings of the ‘A’ Recirculation Pump Motor Shows Indications of Moisture Intrusion; Dated February 15, 2016
- CR 2015–05243; Unexpected Half Scram Received; Dated April 15, 2015
- CR 2015–05355; Re-performance of Steps Required for SVI–B21–T2223; Dated April 16, 2015
- CR 2016–02180; Erroneous Thermal Limit Calculation Following Recirc Pump Trip; Dated February 15, 2016
- CR 2016–02093; NOBP–TR–1151 Crew Performance Critique – Loss of Bus EH11, Loss of Shutdown Cooling; Dated February 12, 2016
- CR 2016–01866; Manual Reactor SCRAM Based on Suppression Pool Temperature of 95 Degrees F due to Open SRVs SCRAM 1–16–02; Dated February 8, 2016
- Drawing 208–0206–00046; Metal-clad Switchgear (15kV & 5kV) 4.16kV Bus EH11 Under Voltage and Potential Circuits; Revision Z
- CR 2016–02128; Safety Relief Valve 1B21F0047H Seat Leakage; Dated February 13, 2016
- CR 2016–02088; EH11 Fuse had Inconsistent Resistance Readings; Dated February 12, 2016
- CR 2016–02158 CR Tracking Completion of ODMI for Operation with Minor SRV 1B21F0047H Leakage; Dated February 14, 2016
- CR 2016–02140; Switch Yard Breaker S611 Failed to Close; Dated February 13, 2016
- CR 2016–02011; 1B21N0062B Pressure Spike for 10 Milliseconds; Dated February 11, 2016
- CR 2016–02070; Defect Found with the ‘A’ Phase Secondary PT Fuse; Dated February 12, 2016
- CR 2016–02056; Abnormal Indications on RCIS Following Power Restoration to EH11; Dated February 12, 2016
- CR 2016–02060; Div 1 DC Normal Charger did not Respond as Expected; Dated February 12, 2016
- CR 2016–02073; NRC Identified: Questions Regarding the 2/8/16 Plant Scram; Dated February 12, 2016
- CR 2016–02059; Not all Valves Closed During Expected BOP Isolation; February 12, 2016
- Perry Reactivity Plan/Power Profile for Start-up from February 2016 Forced Outage; Dated February14, 2016
- NOP–OP–1002; Conduct of Operations; Revision 11
- OAI–1703; Hardcards; Revision 25
- IOI–3; Power Changes; Revision 58
- IOI–3; Power Changes; Revision 59
- IOI–3; Power Changes; Revision 60
- IOI–3; Power Changes; Revision 61
- IOI–4; Shutdown; Revision 22
- CR 2016–01071; Reactor Recirc Loop ‘A’ Pump Discharge Valve Vent Line Leakage; Dated January 24, 2016
- Drawing 304–0601–00103; Reactor Recirculation Valve Flow Control System, Reactor Building; Revision J
- ECP 13–0818–002; Modification of Vent Appendage Attached to 1B33F0060A in the B33 Reactor Recirculation System (Appendage Replacement); Revision 0
- CR 2016–01063; Reactor SCRAM on RPV Level 8 – SCRAM No 1–16–01; Dated January 24, 2016
- SOI–C34; Feedwater Control System; Revision 34
- NOP–OP–1015; Event Notifications; Revision 2
- CR 2016–04016; NRC ID 2016 SIT: Potential NRC Finding Related to Compliance with Branch Technical Position PSB–1 C.3; Dated March 28, 2016
- Event Notice 51681; Average Power Range Monitoring System Inoperable; Dated January 24, 2016

- CR 2016–03861; NRC ID 2016 SIT: Inaccurate Category Classification of Condition Report 2016–02060; Dated March 24, 2016
- 10CFR50.59 Screening No. 16–00337; APRM Changes to IOI–3; Dated January 27, 2016

1R22 Surveillance Testing
- SVI–C71–T0427; RX Mode Switch Refuel Mode Channel Functional; Dated January 25, 2016
- SVI–P47–T2001A; Control Complex Chilled Water 'A' Pump and Valve Operability Test; Dated February 25, 2016
- NOP–WM–2003; Work Management Surveillance Process; Revision 08
- PDB–C0013; SVI and PTI Availability; Revision 3
- SVI–C51–T0030–G; APRM 'G' Channel Calibration for 1C51–K605G; Dated February 19, 2016
- NOP–OP–1014; Plant Status Control; Revision 04
- NOBP–LP–2601; Human Performance Program; Revision 10

1EP6 Drill Evaluation
- OTLC–3058201606 PY–SGC2–Cycle 6 2016 Evaluated Scenario C2; Revision 0

4OA1 Performance Indicator Verification
- NOBP–LP–4012; NRC Performance Indicators; Revision 5
- NOBP–LP–4012–01, Revision 2; Unplanned Scrams per 7,000 Critical Hours; January 2015 through December 2015
- NOBP–LP–4012–02, Revision 2; Unplanned Scrams with Complications (USwC); January 2015 through December2015
- NOBP–LP–4012–03, Revision 2; Unplanned Power Changes per 7,000 Critical Hours; January 2015 through December2015

4OA2 Problem Identification and Resolution
- CR 2016–02721; Unacceptable Preconditioning Identified in Cycle 16, Period 03, Week 11 Schedule; Dated February 26, 2016
- CR 2016–02690; During the Development of ECP 15-0545 it was Identified that SMRF 99–5046 Drawing Updates did not Identify Unit 2 High Pressure Core Spray (HPCS) Electrical Room CC 620–10 Walls as a F–3B Barrier; Dated February 26, 2016
- NOBP–LP–4014; Managing Regulatory Interface; Revision 6
- CR 2016–00503; NRC NCV – Inadequate Procedure for Operations of DG Building Ventilation System; Dated January 13, 2016
- CR 2015–08931; Upper Containment Airlock Outer Door Leakage not Reported to Control Room in a Timely Manner; July 1, 2015
- CR 2015–09827; ERO Pager Test not Successfully Completed; Dated July 21, 2015
- CR 2015–09852; Unsecured Safeguard Cabinet; Dated July 21, 2015
- CR 2015–10109; 2015 NRC PI&R Inspection: Less than Adequate use-as-is Justification for CR 2013–09353 and ECP 13–0439; Dated July 28, 2015
- CR 2015–10501; Unexpected RCIC Isolation during SVI–E31–T5395B; Dated August 5, 2015
- CR 2015–10559; 2015 NRC PI&R Inspection: Question Raised Regarding the Classification of the RPV Leak Detection Line; Dated August 7, 2015
- CR 2015–10777; Flashing Landed on U2 Startup Transformer; Dated August 12, 2015
- CR 2015–11288; Potential Cognitive Trend in Safety and Human Performance at Perry; Dated August 26, 2015
- CR 2015–11906; Increasing Trend in Reactor Water Co–60 and Mn–56; Dated September 9, 2015

- CR 2015–12258; Gagged ECC Valves Require Greater than 5 Rem for Required Post-accident Manual Operator Action; Dated September 17, 2015
- CR 2015–12550; FO–SA–2015–0066, Failure to Properly Follow Process for Portable Media and Mobile Devices (PMMD); Dated September 24, 2015
- CR 2015–12627; Year to Date Search Results Indicate 10 Clearance Request Errors; Dated September 25, 2015
- CR 2015–12755; Radio Channel 1 Power Supply Failed; Dated September 26, 2015
- CR 2015–12952; 2015 NRC Fire Protection Inspection – Barrier Penetration Inspections; September 30, 2015
- CR 2015–13093; On 9/21/2015 Semi-Annual Safety Culture Monitoring Panel Meeting, Trait Personal Accountability (PA) and Trait Work Processes (WP) was Rated as an Area in Need of Improvement; Dated October 1, 2015
- CR 2015–13299; 2015 NRC Fire Protection Inspection – Potential Non-Conformance with Appendix R due to Hot Shutdown; Dated October 5, 2015
- CR 2015–13494; Potential Trend Identified in Safety Hazard Awareness for Fire Protection; Dated October 7, 2015
- CR 2015–14788; CNRB ID: Condition Reports 2015–01437 and 2015–01436, from the February 2015 CNRb Executive Summary, were Closed Prior to Action Being Taken; Dated October 29, 2015
- CR 2015–15089; Received HPCS and RCIC Suction Swap during RHR 'B' Recovery; Dated November 4, 2015
- CR 2015–15368; NRC ID: NRC Resident Inspector Identified Preconditioning Concern for Division 2 Diesel Generator; Dated November 10, 2015
- CR 2015–16098; Clearance Event – MH 23 Circuit 1D17R68X Removed; Dated November 30, 2015
- CR 2015–16225; NRC Cyber Security Inspection – Inadequate Logical and Physical Controls on Portable Device; Dated December 2, 2015
- CR 2015–16548; NRC NOV: Radiation Protection Manager Qualifications; Dated December 9, 2015
- CR 2016–03861; NRC ID 2016 SIT: Inaccurate Category Classification of Condition Report 2016–02060; Dated March 24, 2016
- CR 2016-02060; Div 1 DC Normal Charger did not Respond as Expected; Dated February 12, 2016
- Maintenance Rule Failure Review Form for DC Battery Charger Failure on February 20, 2016, part of CR 2016–02441
- CR 2016–02441; Electrical Transient during Shift from Charger EFD–12–A to Charger EFD–1A; Dated February 20, 2016
- CA–2015–15368–001; Preconditioning Corrective Action; Dated December 4, 2015

LIST OF ACRONYMS USED

ADAMS	Agencywide Document Access Management System
APRM	Average Power Range Monitor
CAP	Corrective Action Program
CFR	*Code of Federal Regulations*
CR	Condition Report
gpm	Gallons Per Minute
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IR	Inspection Report
LOCA	Loss of Coolant Accident
MFP	Motor-Drive Reactor Feed Pump
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PI	Performance Indicator
QC	Quality Control
PMT	Post Maintenance Testing
RCS	Reactor Coolant System
RFPT	Turbine-Driven Reactor Feed Pump
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SRV	Safety Relief Valve
TS	Technical Specification
USAR	Updated Safety Analysis Report
WO	Work Order

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Sincerely,

/RA/

Billy Dickson, Chief
Branch 5
Division of Reactor Projects

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