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

November 6, 2015

EA-15-203

Mr. Bryan C. Hanson
Senior VP, Exelon Generation Company, LLC
President and CNO, Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION, UNIT 1 – NRC INTEGRATED INSPECTION REPORT 05000461/2015003 AND EXERCISE OF ENFORCEMENT DISCRETION

Dear Mr. Hanson:

On September 30, 2015, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station, Unit 1. The enclosed report documents the results of this inspection, which were discussed on October 15, 2015, with Mr. D. Kemper and other members of your staff. The inspectors documented the results of this inspection in the enclosed inspection report.

Based on the results of this inspection, four NRC-identified findings of very low safety significance were identified. The findings involved a violation of NRC requirements. However, because of their very low safety significance, and because the issues were entered into your Corrective Action Program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section 2.3.2 of the NRC Enforcement Policy.

Separately, a violation involving a failure to establish secondary containment operability during operations with the potential to drain the reactor vessel (OPDRV) was identified during the last refueling outage. Specifically, from April 27, 2015, to May 2, 2015, and from May 6, 2015, to May 12, 2015, while all other Technical Specifications (TSs) were met, Clinton Power Station conducted several OPDRVs without establishing secondary containment operability, which is a violation of TS 3.6.4.1. The NRC issued Enforcement Guidance Memorandum (EGM) 11-003, "Enforcement Guidance Memorandum on Dispositioning Boiling Water Reactor Licensee Noncompliance with Technical Specification Containment Requirements During Operations with a Potential for Draining the Reactor Vessel," Revision 2, on December 13, 2013, allowing for the exercise of enforcement discretion for such OPDRV-related TS violations, when certain criteria are met. The NRC concluded that, for this specific period, Clinton Power Station met the EGM criteria. Therefore, I have been authorized, after consultation with the Director, Office of Enforcement, and the Regional Administrator, to exercise enforcement discretion and refrain from issuing enforcement for the violation.

If you contest the subject or severity of these NCVs, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U.S.

Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Clinton Power Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Clinton Power Station.

In accordance with Title 10, *Code of Federal Regulations* (CFR), Part 2.390, "Public Inspections, Exemptions, Requests for Withholding," of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (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/

Patrick Louden, Director
Division of Reactor Projects

Docket No. 50–461
License No. NPF–62

Enclosure:
Inspection Report 05000461/2015003

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50–461
License No:	NPF–62
Report No:	05000461/2015003
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, IL
Dates:	July 1 through September 30, 2015
Inspectors:	W. Schaup, Senior Resident Inspector E. Sanchez-Santiago, Resident Inspector A. Shaikh, Reactor Inspector J. Bozga, Reactor Inspector M. Jones, Reactor Inspector C. Phillips, Project Engineer S. Mischke, Resident Inspector, Illinois Emergency Management Agency
Approved by:	P. Louden, Director Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

Inspection Report 05000461/2015003, 07/01/2015 – 09/30/2015, Clinton Power Station, Unit 1; Operability Determinations and Functionality Assessments, Plant Modifications, and Follow-Up of Events and Notices of Enforcement Discretion.

This report covers a 3-month period of inspection by the resident inspectors and announced baseline inspections by regional inspectors. Four Green findings were identified by the inspectors. The findings were Non-Cited Violations (NCVs) of U.S. Nuclear Regulatory Commission (NRC) regulations. 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," effective date 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" Revision 5, dated February 2014.

Cornerstone: Initiating Events

- Green. The inspectors identified a finding of very low safety significance, and an associated NCV of Title 10, *Code of Federal Regulations* (CFR), Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to maintain control room cabinet doors in a seismically analyzed condition in accordance with procedure Clinton Power Station (CPS) 1014.11, "6900/4160/480v Switchgear/ Circuit Breaker Operability Program," Revision 5a. Specifically, the licensee failed to maintain control room cabinet doors in seismically qualified positions while performing maintenance or trouble shooting activities due to leaving the doors open and unattended. The licensee documented the issue in the Corrective Action Program (CAP) as action request (AR) 02518477. The licensee revised the station procedure to ensure control room cabinet doors either remain latched closed or be completely removed when unattended and had issued a standing order to ensure the requirements were reinforced.

  The performance deficiency was more than minor because it was associated with the configuration control performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations and was therefore a finding. Specifically, leaving the doors in a seismically unanalyzed condition could challenge critical safety functions during a seismic event. The finding was screened against the Initiating Events cornerstone and determined to be of very low safety significance because it did not result in exceeding the reactor coolant system leak rate for a small loss of coolant accident, did not cause a reactor trip, did not involve the complete or partial loss of a support system that contributed to the likelihood of, or caused, an initiating event and did not affect mitigation equipment. This finding had a cross-cutting aspect of resources in the area of human performance because the licensee failed to ensure the personnel performing maintenance and troubleshooting had adequate documentation in written work instructions to maintain control room cabinets in seismically analyzed conditions. [H.1] (Section 1R15)

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance, and an associated NCV of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," for the licensee's failure to implement and comply with procedure CPS 1019.05, "Transient Equipment/Materials," Revision 23. Specifically, the inspectors identified equipment and materials improperly staged, improperly secured, or placed in areas without engineering evaluations. The licensee documented the issue in the CAP as AR 02507167 and AR 02529227. The licensee subsequently removed the items identified to restore compliance with the procedure.

  The inspectors determined this issue was more than minor because if left uncorrected, it had the potential to lead to a more significant safety concern. Specifically, transient equipment and material in proximity of safety related components has the potential of impacting these components during a seismic event, potentially rendering them unable to fulfill their safety function. The finding was determined to be of very low safety significance because it did not represent a loss of system or function, did not represent an actual loss of function of at least a single train for greater than its Technical Specifications (TSs) allowed outage time, and did not represent an actual loss of one or more non-TS trains of equipment designated as high safety-significant in the licensee's Maintenance Rule Program. This finding had a cross-cutting aspect of field presence in the area of human performance because the licensee did not perform in field observations, coaching and reinforcement of standards and expectations in the identified areas after various examples of material placement issues were identified. [H.2] (Section 1R15)

Cornerstone: Barrier Integrity

- Severity Level IV. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," on December 18, 2014, for the licensee's failure to provide a written evaluation showing that a change to the secondary containment did not require a license amendment. Specifically, the licensee eliminated the tornado wind and tornado missile loading conditions from the fuel building railroad airlock enclosure walls, roof and associated outer door Seismic Category I design requirements. However, the licensee failed to provide a written evaluation describing why the change would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component important to safety. The licensee documented the issue in the CAP as AR 02534694. Corrective actions included complying with TS anytime the inner railroad bay door is opened by entering the applicable action statements, evaluating weather conditions and impact to plant risk and establishing the necessary mitigating actions required prior to opening the door.

  The inspectors determined that the licensee's failure to provide a written evaluation that documented the basis for determining that the change to the secondary containment did not require a license amendment was a performance deficiency. The performance deficiency was determined to be more than minor because it was associated with the design control attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system and containment) protect the public from radionuclide releases caused by accidents or events. In addition, the associated violation was determined to be more than minor because the inspectors could not reasonably determine if the changes to secondary containment would have required prior NRC approval. Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the significance determination process (SDP) because they are considered to be violations that potentially impede or impact the regulatory process. However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the finding was screened against the Barrier Integrity cornerstone, and determined to be of very low safety significance because the finding did not represent a degradation only of the radiological barrier function for the standby gas treatment system nor did it represent a degradation of the function of the control room against smoke or toxic atmosphere. The inspectors determined this finding has a cross-cutting aspect of procedure adherence in the area of human performance because the licensee failed to follow the 50.59 regulatory process as defined in procedure LS-AA-104-1000, "50.59 Resource Manual," Revision 9. [H.8] (Section 1R18)

- Green. The inspectors identified a finding, and an associated NCV of TS 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation," and TS 3.3.6.2, "Secondary Containment Isolation Instrumentation," due to the failure to enter the appropriate TS action statement and take the required actions when the containment radiation monitoring instrumentation was inoperable during operations with the potential to drain the reactor vessel (OPDRVs). Specifically, with the containment ventilation dampers closed, the containment radiation monitoring instrumentation was unable to perform its safety function of sending a containment isolation signal to various equipment if elevated containment radiation levels occurred during OPDRVs. The licensee documented the issue in the CAP as AR 02566708. The inspectors identified this issue after the maintenance on the containment ventilation system and the OPDRVs were completed. Therefore, the TS non-compliance was no longer in effect.

  The inspectors determined that the failure to enter TS 3.3.6.1 and TS 3.3.6.2 when the containment radiation monitoring instrumentation was not able to perform its safety function during an OPDRV was a performance deficiency. The performance deficiency was more than minor because it was associated with the systems, structures and components and barrier performance attributes of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events, and is therefore a finding. Based on a detailed risk evaluation, the finding screened as very low safety significance because the reactor water level was confirmed to be greater than the minimum level required for movement of irradiated fuel assemblies (i.e., greater than 22'8" above the flange) during all OPDRV time windows. This finding had a cross-cutting aspect of conservative bias in the area of human performance because the licensee relied solely on the successful completion of the surveillance requirements to determine the radiation monitor instrumentation was operable rather than considering the impact the closed dampers would have on their ability to fulfill their safety function. [H.14] (Section 4OA3)

Licensee-Identified Violations

None.

REPORT DETAILS

Summary of Plant Status

The unit was operated at or near full power during the inspection period with the following exception:

- On September 6, 2015, power was reduced to approximately 80 percent to perform a rod line adjustment and to support turbine valve testing. The unit was returned to full power the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04Q)

.1 Quarterly Partial System Walkdowns

a. Inspection Scope

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

- Fuel pool cooling train "B" while train "A" was undergoing maintenance; and
- Residual heat removal train "C" after maintenance.

The inspectors selected these systems based on their risk significance relative to the Reactor Safety cornerstones. The inspectors reviewed operating procedures, system diagrams, Technical Specification (TS) requirements, and the impact of ongoing work activities on redundant trains of equipment. The inspectors verified that conditions did not exist that could have rendered the systems incapable of performing their intended functions. The inspectors also walked down accessible portions of the systems to verify system components were aligned correctly and available as necessary.

In addition, the inspectors verified that equipment alignment problems were entered into the licensee's Corrective Action Program (CAP) with the appropriate characterization and significance. Selected ARs were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

These activities constituted two partial system walkdown samples as defined in Inspection Procedure (IP) 71111.04–01.

b. Findings

No findings were identified.

.2 Semi-Annual Complete System Walkdown

a. Inspection Scope

From August 17, 2015, through August 20, 2015, the inspectors performed a complete system alignment inspection of the control rod drive system to verify the functional capability of the system. This system was selected because it was considered both safety-significant and risk-significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review mechanical and electrical equipment lineups; electrical power availability; system pressure and temperature indications, as appropriate; component labeling; component lubrication; component and equipment cooling; hangers and supports; operability of support systems; and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding work orders was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the CAP database to ensure that system equipment alignment problems were being identified and appropriately resolved.

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

b. Findings

No findings were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (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 zone D–2, diesel generator building division 1 diesel fuel tank room - elevation 712';
- Fire zone A–3a, residual heat removal B equipment rooms - elevation 707';
- Fire zone T–1m, turbine deck - elevation 800'; and
- Fire zone F–1p, fuel pools and general access – elevations 712' and 737'.

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

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

b. Findings

No findings were identified.

1R06 Flooding (71111.06)

.1 Internal Flooding

a. Inspection Scope

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

- Control building 781', division 1 cable spreading room.

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

b. Findings

No findings were identified.

.2 Underground Vaults

a. Inspection Scope

The inspectors selected underground bunkers/manholes subject to flooding that contained cables whose failure could disable risk-significant equipment. The inspectors determined that the cables were not submerged, that splices were intact, and that appropriate cable support structures were in place. In those areas where dewatering devices were used, such as a sump pump, the device was operable and level alarm circuits were set appropriately to ensure that the cables would not be submerged. In those areas without dewatering devices, the inspectors verified that drainage of the area was available, or that the cables were qualified for submergence conditions. The inspectors also reviewed the licensee's corrective action documents with respect to past submerged cable issues identified in the corrective action program to verify the adequacy of the corrective actions. The inspectors performed a walkdown of the following safety related underground bunkers/manholes subject to flooding:

- Cable vault OSHA–1B, division 2 shutdown service water (SX); and
- Cable vault OSHA–1C, division 3 SX.

This inspection constituted one underground vault sample as defined in IP 71111.06–05.

b. Findings

No findings were identified.

1R07 Annual Heat Sink Performance (71111.07)

.1 Heat Sink Performance

a. Inspection Scope

The inspectors reviewed the licensee's test report from the division 3 SX system testing to verify that potential deficiencies did not mask the licensee's ability to detect degraded performance, to identify that any common cause issues that had the potential to increase risk, and to ensure that the licensee was adequately addressing problems that could result in initiating events that would cause an increase in risk. The inspectors reviewed the licensee's observations as compared against acceptance criteria, the correlation of scheduled testing and the frequency of testing, and the impact of instrument inaccuracies on test results. The inspectors also verified that test acceptance criteria considered differences between test conditions, design conditions, and testing conditions.

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

b. Findings

No findings were identified.

1R08 Inservice Inspection Activities (71111.08)

From April 27, 2015, through July 6, 2015, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system, risk-significant piping and components, and containment systems.

The reviews described in Sections 1R08.1 and 1R08.5 below, count as one inspection sample as defined in IP 71111.08.

.1 Piping Systems Inservice Inspection

a. Inspection Scope

The inspectors reviewed records of the following Non-Destructive Examinations (NDEs) required by the American Society of Mechanical Engineers (ASME) code, Section XI, and/or Title 10, *Code of Federal Regulations* (CFR), Part 50.55a, to evaluate compliance with the ASME code, Section XI, and Section V requirements, and if any indications and defects were detected, to determine whether these were dispositioned in accordance with the ASME code or U.S. Nuclear Regulatory Commission (NRC)-approved alternative requirement:

- VT (Visual Testing)–3, visual examination NDE report for component supports, attachments and interiors of reactor vessels, 1RH07074X;
- VT–3, visual examination NDE report for component supports, attachments and interiors of reactor vessels, 1SX22003A;
- VT–3, visual examination NDE report for component supports, attachments and interiors of reactor vessels, 1SX01004R; and
- VT–2, visual examination of blind coupling pressure boundary weld on stand-by liquid control line, 1SC27A-2.

During non-destructive surface and volumetric examinations performed since the previous refueling outage, the licensee had not identified any recordable indications. Therefore, no NRC review was completed for this inspection procedure attribute.

The inspectors reviewed records of the following risk-significant pressure boundary ASME code, Section XI, class 2, welds fabricated since the beginning of the last refueling outage to determine if the licensee: followed the welding procedure; applied appropriate weld filler material; and implemented the applicable Section XI or construction code NDEs and acceptance criteria. Additionally, the inspectors reviewed the welding procedure specification and supporting weld procedure qualification records to determine if the weld procedure was qualified in accordance with the requirements of construction code and the ASME code, Section IX:

- Class 2 – replace valve 1E22-F006 – high pressure core spray water leg pump discharge stop-check valve (Work Order 01312496); and
- Class 2 – install blind coupling on stand-by liquid control line 1SC27A-2.

b. Findings

No findings were identified.

.2 Identification and Resolution of Problems

a. Inspection Scope

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

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

The inspectors performed these reviews to evaluate compliance with 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," requirements.

b. Findings

No findings were identified.

1R11 Licensed Operator Requalification Program (71111.11)

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

a. Inspection Scope

On August 19, 2015, the inspectors observed a crew of licensed operators in the simulator during licensed operator requalification training to verify that operator performance was adequate, evaluators were identifying and documenting crew performance problems, and training was being conducted in accordance with licensee procedures. The inspectors evaluated the following areas:

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

The crew's performance in these areas was compared to pre-established operator action expectations and successful critical task completion requirements.

This inspection constituted one quarterly Licensed Operator Requalification Program simulator sample as defined in IP 71111.11

b. Findings

No findings were identified.

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

a. Inspection Scope

On September 6, 2015, the inspectors observed the control room operators perform a down power of the unit to about 80 percent power and perform turbine stop and control valve testing. 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;
- 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.

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

b. Findings

No findings were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations

a. Inspection Scope

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

- 10 CFR 50.65(a)(3) periodic assessment of the Maintenance Rule Program.

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

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

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the system. In addition, the inspectors verified maintenance effectiveness issues were entered into the CAP with the appropriate significance characterization.

This inspection constituted one 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)

.1 Maintenance Risk Assessments and Emergent Work Control

a. Inspection Scope

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

- Planned yellow risk during reactor core isolation cooling valve maintenance;
- Planned yellow risk during division 1 automatic depressurization system filter change out ;
- Planned green risk during residual heat removal pump “C” maintenance; and
- Impact on plant risk during a geomagnetic flare.

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

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

b. Findings

No findings were identified.

1R15 Operability Determinations and Functional Assessments (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- Action Request (AR) 02547254 – Reactor Recirculation “B” Flow Control Valve Locked Out on High Temperature;
- AR 02512414 – NRC Observation of Main Control Room Back Panel Door Control;
- AR 02523496 – 0FP01PB (fire pump “B”) Data from 9071.02 Requires Evaluation;
- AR 02507167 – Non-compliance with CPS 1019.05;
- Engineering Change Request 419711 – Over Pressurization of Drywell Seal during seal leak rate test; and
- AR 02541236 – Part 21: NAMCO Limit Switches (EA 170/EA 180).

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 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.

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

b. Findings

(1) Failure to Follow Procedure Leaves Unattended Control Room Cabinet Doors in Seismically Unanalyzed Condition

Introduction. The inspectors identified a finding of very low safety significance, and an associated Non-Cited Violation (NCV) of 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to maintain control room cabinet doors in a seismically analyzed position in accordance with procedure Clinton Power Station (CPS) 1014.11, “6900/4160/480v Switchgear/Circuit Breaker Operability Program,” Revision 5a. Specifically, the licensee failed to maintain control room cabinet doors in seismically qualified positions due to leaving the doors open and unattended during maintenance activities.

Description. While performing a control room tour on May 1, 2015, the inspectors identified a power cord that had been routed into an unattended control room cabinet that required leaving the door ajar. During the same tour, the inspectors identified another unattended control room cabinet door open with various pieces of testing equipment attached to the cabinet internals. The inspectors asked the shift manager if the control room cabinet doors had seismic qualifications and if the current positions for the doors had been analyzed. The issue was documented in AR 02494259. The control room operations staff had the power cord rerouted and ensured the other door was not left unattended.

In a follow up discussion with engineering, it was determined that the safety related control room cabinets were seismically qualified with the access doors closed and secured (e.g., latched). The qualification was later extended to address the qualification of the cabinets with the doors fully removed and stored in a secure location.

During a control room tour on June 5, 2015, the inspectors identified a power cord that had been routed into an unattended control room cabinet. The routing of the cord required leaving the door ajar. The inspectors questioned the shift manager whether the door was in a seismically unanalyzed condition. This issue was documented in AR 02512414. The licensee determined the door condition did not meet the instructions in procedure CPS 1014.11, “6900/4160/480v Switchgear/Circuit Breaker Operability Program,” that stated the cabinet was only seismically analyzed for the condition of the door being fully removed and stored in a secure location. The licensee rerouted the cord and secured the door.

During a control room tour on June 23, 2015, the inspectors identified that the division 1 nuclear system protection system control room cabinet door was open and unattended with various pieces of maintenance and testing equipment attached to cabinet internals to support troubleshooting activities. The inspectors contacted the shift manager and pointed out that the door was not in a seismically analyzed condition as required by procedure CPS 1014.11. The issue was documented in AR 02518477. The work group was contacted, the door was attended, an immediate operability determination was performed and the work group performed a stand down on the occurrence.

During discussions with personnel and during a review of supporting work documents, the inspectors determined that the work instructions provided to personnel performing the work did not reference procedure CPS 1014.11, nor did they provide any instruction on what positions the control room cabinet doors could be left to maintain seismic qualification. Additionally, personnel performing the work were unaware that the doors had seismically qualified positions.

Analysis. The inspectors determined that the failure to maintain the control room cabinet doors in a seismically analyzed condition, in accordance with procedure CPS 1014.11, “6900/4160/480v Switchgear/Circuit Breaker Operability Program,” Revision 5a, was a performance deficiency. Specifically, the licensee failed to maintain control room cabinet doors in seismically qualified positions while performing maintenance or trouble shooting activities due to leaving the doors open and unattended. The performance deficiency was more than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the configuration control performance attribute of the Initiating Events cornerstone and adversely affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations and is therefore a finding. Specifically, leaving the cabinet doors in a seismically unanalyzed condition could challenge critical safety functions during a seismic event. Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process for Findings at Power,” issued June 19, 2012, the finding was screened against the Initiating Events cornerstone, and determined to be of very low safety significance (Green) because the finding did not result in exceeding the reactor coolant system leak rate for a small loss of coolant accident, cause a reactor trip, involve the complete or partial loss of a support system that contributes to the likelihood of, or caused, an initiating event and did not affect mitigation equipment.

The inspectors determined this finding had a cross-cutting aspect of resources in the area of human performance where leaders ensure that personnel, equipment, procedures and other resources are available and adequate to support nuclear safety. Specifically, the licensee failed to ensure the personnel performing maintenance and troubleshooting had adequate documentation in written work instructions to maintain control room cabinets in seismically analyzed conditions. [H.1]

Enforcement. Title 10 CFR Part 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented procedures of a type appropriate to the circumstance and be accomplished in accordance with these procedures. The licensee established CPS 1014.11, “6900/4160/480v Switchgear/Circuit Breaker Operability Program,” Revision 5a, as the implementing procedure for maintaining control room cabinet doors in seismically qualified conditions. Procedure CPS 1014.11, Step 6.2, states, in part, the seismic qualification of the affected main control room cabinet back panels is maintained fully closed or with the door fully removed and stored in a secure location.

Contrary to the above, on May 1, 2015, June 5, 2015, and June 23, 2015, the licensee failed to perform activities affecting quality, in accordance with procedure CPS 1014.11, Step 6.2, when performing maintenance or troubleshooting activities associated with control room cabinet doors. Specifically, during these activities, station personnel left the associated control room cabinet doors in the open position instead of removing the door as required by procedure to maintain seismic qualification of the cabinet. The licensee has revised the station procedure to ensure control room cabinet doors either remain latched closed or completely removed when unattended and has issued a standing order to ensure the requirements are reinforced. Because this violation was of very low safety significance and was entered into the licensee's CAP as AR 02518477, this violation is being treated as NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2015003-01, Failure to Follow Procedure Leaves Control Room Cabinet Doors Unattended in Seismically Unanalyzed Condition)**

(2) Failure to Implement and Comply with Transient Equipment/Materials Program

Introduction. The inspectors identified a finding of very low safety significance and an associated NCV of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" for the licensee's failure to implement and comply with procedure CPS 1019.05, "Transient Equipment/Materials," Revision 23, to ensure that transient equipment and materials are controlled so there is no impact to safe operation of plant equipment. Specifically, the inspectors identified equipment and materials improperly staged, improperly secured or in areas without engineering evaluations.

Description. During multiple routine walkdowns of the plant in early 2014, the inspectors noted that an approved storage area at the auxiliary building 707' elevation just outside the reactor core isolation cooling (RCIC) room was accumulating transient materials (e.g., tele-towers, large wheeled tool boxes, scaffolding materials, 55-gallon drums full of lead shielding). Some of these items were in close proximity to the RCIC pump room vent panel, 1PL62J, which is a seismically-qualified panel. Additionally, there was a Cotterman maxi-lift stored between seismically-qualified panels 1PL61JB, "Residual Heat Removal Pump and Heat Exchanger Room 1B panel", and 1PL61JC, "Residual Heat Removal Pump Room 1C Vent Panel." This item was outside the approved storage area. The inspectors questioned the licensee on the potential impact of these items on the ventilation panels during a seismic event. The issue was documented in AR 01630607 on March 7, 2014. Licensee actions included evaluations for past operability, removing items from the vicinity of the seismically-qualified panels, clearing out the storage area, and marking the "restricted" areas around the panels in question.

During a routine plant walk down on June 5, 2014, the inspectors noted a portable Tele-tower was stored on the 755' elevation of containment behind air handling Unit 1WO05SC. This area of containment is in the suppression pool swell zone. Although the tele-tower appeared to be adequately restrained with nylon strapping, a technical evaluation had not been performed to determine if the strapping was adequate to secure the tele-tower in accordance with station procedures. AR 01668804 was written to document the issue. The licensee subsequently removed the tele-tower from containment.

During a routine plant walk down on April 22, 2015, the inspectors noted a wheeled cart containing numerous scaffolding poles was stored on the 707' elevation of the auxiliary building near 1PL62J, “Reactor Core Isolation Cooling Room Ventilation Panel.” Signs on the panel, hung in accordance with station procedures, stated that items were not to be stored/staged within 3 feet of the panel. Initially the licensee did not document the issue in the CAP, did not perform a past operability determination, nor perform evaluations required by station procedures. AR 02507167 was written to document the issue. The licensee subsequently removed the cart from the area.

During a routine plant walk down on May 21, 2015, the inspectors noted a large wheeled toolbox was stored next to the RCIC room ventilation panel. Signs on the panel, hung in accordance with station procedures, stated that items were not to be stored/staged within 3 feet of the panel. Initially the licensee did not document the issue in the CAP, perform a past operability determination or perform evaluations required by station procedures. AR 02507167 was written to document the issue. The licensee subsequently removed the toolbox from the area.

During a routine plant walk down on July 16, 2015, the inspectors identified a tele-tower staged in the division 3 diesel generator (DG) room. The tele-tower was placed in proximity of SX piping. This piping is safety related and its purpose is to support the operation of the DG. Staging the tele-tower in proximity to this piping was contrary to station procedures. This was documented in AR 02529227. The tele-tower was removed from the area and an evaluation was completed determining that there was no impact to the safety related equipment and the licensee would revise procedures to clarify the requirements for tele-towers.

The inspectors reviewed procedure CPS 1019.05, “Transient Equipment and Materials,” Revision 22, which is used to control the placement of transient equipment near safety-related equipment within the plant. Step 8.5.1 states that stable items that are not “in-use” shall not be set within 3 feet plus their height of adjacent equipment or ledges unless positively secured. Step 8.5.2 states that unstable items that are not “in-use” shall be secured by appropriate means above their center of gravity to prevent falling and damaging adjacent equipment. Step 8.5.6 states that items stored outside “Approved Storage Areas” require engineering technical evaluation.

The inspectors determined that the items discovered during routine walk downs in the plant represented five examples of the licensee failing to implement or comply with the requirements of procedure CPS 1019.05. Therefore, based on the multiple examples listed above, the inspectors determined that the failures to be of a routine nature and indicative of a programmatic failure to ensure that transient equipment and materials were controlled so there were no impact to safe operation of plant equipment as required per the transient equipment and material process.

Analysis. The inspectors determined the licensee’s failure to implement and comply with procedure CPS 1019.05 when staging or storing equipment and materials in the plant was a performance deficiency. The performance deficiency was determined to be more than minor in accordance with IMC 0612, “Power Reactor Inspection Reports,” Appendix B, “Issue Screening,” dated September 7, 2012, because if left uncorrected it had the potential to lead to a more significant safety concern. Specifically, transient equipment and material in proximity of safety related components has the potential of impacting these components during a seismic event, potentially rendering them unable to perform their safety function. The performance deficiency is also associated with the protection against external factors attribute of the Mitigating Systems cornerstone, and adversely affected the cornerstone objective to ensure the availability, reliability and capability of systems that response to initiating events to prevent undesirable consequences, and is therefore a finding. Using IMC 0609, Attachment 4, “Initial Characterization of Findings,” and Appendix A, “The Significance Determination Process for Findings at Power,” issued June 19, 2012, the finding was screened against the Mitigating Systems cornerstone and determined to be of very low safety significance (Green) because the finding did not represent a loss of system or function, did not represent an actual loss of function of at least a single train for greater than its TS allowed outage time, and did not represent an actual loss of one or more not TS trains of equipment designated as high safety-significant in accordance with the licensee’s Maintenance Rule Program.

The inspectors determined this finding had a cross-cutting aspect of field presence in the area of human performance where leaders are commonly seen in the work areas of the plant observing, coaching, reinforcing standards and expectations. Deviations from standards and expectations are corrected promptly. Specifically, after various examples of material placement being an issue, the licensee did not perform in field observations, coaching and reinforcement of standards and expectations in the identified areas. [H.2]

Enforcement. Title 10 CFR Part 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” requires, in part, that activities affecting quality be prescribed by documented procedures of a type appropriate to the circumstances and be accomplished in accordance with these procedures. The licensee established procedure CPS 1019.05 “Transient Equipment/Materials,” Revision 22, as the implementing procedure for controlling the placement of transient equipment within the plant, an activity affecting quality.

Procedure CPS 1019.05, Step 8.5.1, states that stable items that are not “in-use” shall not be set within 3 feet plus their height of adjacent equipment or ledges unless positively secured. Step 8.5.2 states that unstable items that are not “in-use’ shall be secured by appropriate means above their center of gravity to prevent falling and damaging adjacent equipment. Step 8.5.6 states that items stored outside “Approved Storage Areas” require engineering technical evaluation.

Contrary to the above, around March 7, 2014, and on June 5, 2014, April 22, 2015, May 21, 2015, and July 16, 2015, the licensee failed to implement and comply with procedure CPS 1019.05. Specifically, the licensee failed to comply with Step 8.5.1 by storing equipment within 3 feet plus their height of various safety related equipment including the RCIC ventilation panel and service water piping serving the DG. The licensee also failed to comply with Step 8.5.2 to ensure that the equipment was secured by appropriate means and Step 8.5.6 to ensure the equipment outside of approved storage areas had an engineering evaluation. In each instance, the licensee removed the equipment from the area to restore compliance and additionally performed a site wide communication to reinforce the procedure requirements. Because this violation was of very low safety significance, and was entered into the licensee’s CAP as AR 02507167 and AR 02529227, this violation is being treated as NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/2015003-02, Failure to Implement and Comply with Transient Equipment/Materials Program)**

1R18 Plant Modifications (71111.18)

.1 Plant Modifications

a. Inspection Scope

The inspectors reviewed the following modification:

- Residual Heat Removal (RHR) "A" drain line.

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

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

b. Findings

No findings were identified.

(1) (Closed) Unresolved item 05000461/2015002–01: Fuel Handling Building Railroad Bay Airlock Design and Licensing Basis Issues

a. Inspection Scope

The inspectors reviewed Engineering Change (EC) 395976, "ISFSI-Extend Secondary Containment Boundary to FB Outer Railroad Bay Doors," Revision 0. This engineering change established the boundary of the secondary containment to include the FB railroad bay airlock. The inspectors pursued this issue in order to determine if the licensee had appropriately evaluated and dispositioned the modification in accordance with 10 CFR 50.59, "Changes, Test and Experiments." Based upon the inspectors' questions, the licensee provided additional information on the design and licensing basis of the FB railroad bay airlock that required additional NRC review.

The inspectors reviewed the additional information provided by the licensee and the licensee's procedures and program for implementing 10 CFR 50.59 to permanent plant modifications. The inspectors reviewed the evaluations specific to this issue to understand what the requirements were for the modification and whether the actions taken by the licensee were in compliance with those requirements.

Documents reviewed during this inspection are provided in the Attachment to this report. No inspection sample was credited for this inspection as the sample was credited in a previous integrated inspection report.

b. Findings

Failure to Obtain a License Amendment prior to Making Modifications to Secondary Containment

Introduction. The inspectors identified a Severity Level IV NCV of 10 CFR 50.59(d)(1), "Changes, Tests, and Experiments," for the licensee's failure to provide a written evaluation describing the basis for determining that the change to the secondary containment completed on December 18, 2014, did not require a license amendment. Specifically, the licensee made a change to the secondary containment pursuant to 10 CFR 50.59(c). This change eliminated the tornado wind and tornado missile loading condition from the FB railroad airlock (the enclosure walls and roof), and associated outer door (1SD1-31) Seismic Category I requirements. However, no written evaluation was provided describing the basis for determining that this change would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

Description. The inspectors reviewed EC 395976 that extended the boundary of the secondary containment to include the FB railroad airlock and outer door. The associated 50.59 evaluation, CL-2014-E-033, stated, in part, "USAR Sections 6.2.3.1, 6.2.3.2 and 6.2.3.3 describe the secondary containment boundary, which is shown in USAR Figure 6.2-132. This activity (EC 395976) extended the secondary containment boundary to the FB outer railroad bay airlock doors such that the railroad bay airlock becomes included within the secondary containment whenever the inner railroad bay airlock doors are open. A malfunction of the secondary containment in the railroad bay airlock area (the enclosure walls, roof or outer doors) would have the same result as a malfunction of the secondary containment in any other portion of the boundary. As a result of this EC, the railroad bay airlock meets the USAR described requirements applicable to the secondary containment and consequently this activity does not result in more than a minimal increase in the likelihood of occurrence of a malfunction of the secondary containment as previously evaluated in the USAR.

USAR Section 6.2.3.2 states, in part that, "The secondary containment consists of the fuel building; the portion of the auxiliary building enclosing the ECCS [emergency core cooling system] pump rooms, the RWCU [reactor water cleanup] pump and heat exchanger rooms and the main steam tunnel to S-line, the gas control boundary that encloses the primary containment above the level of the auxiliary and fuel buildings roofs, the radwaste tunnel, the auxiliary building pipe tunnel, main steam isolation valve (MSIV) rooms (for MSIV blowers), auxiliary building floor drain pump room and the gas control boundary extension in the auxiliary building."

USAR Section 6.2.3.1 states, in part that, "The secondary containment structures is of Seismic Category I design…"

USAR Table 3.2-1, Note c, states that, "I = The equipment is constructed in accordance with the seismic requirements of Seismic Category I structures and equipment as described in Section 3.7. All civil structures classified as Seismic Category I structures are designed for the effects of CPS natural phenomena such as tornado, wind loads, external missiles, floods, etc., except the containment gas control boundary building."

Additionally, the inspectors reviewed calculation SDQ15-23DG09, Revision 9C, and identified that FB railroad airlock (the enclosure walls and roof), and associated outer door were not evaluated for the tornado wind loading condition and tornado missile loading condition required by USAR Section 6.2.3. The inspector then reviewed 50.59 evaluation, CL-2014-E-033, to establish where the elimination of the tornado wind loading condition and tornado missile loading condition were evaluated for the FB railroad airlock and associated outer but found that the evaluation was not performed.

The licensee is permitted to make changes to the facility as described in the USAR without prior NRC approval provided that these changes would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety used in establishing the plant design bases. Regulatory Guide 1.187, “Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments,” states that the methods described in Nuclear Energy Institute (NEI) 96-07, “Guidelines for 10 CFR 50.59 Evaluations,” Revision 1, are acceptable to the NRC staff for complying with the provisions of 10 CFR 50.59. NEI 96-07, Revision 1, Section 4.3.2, addresses “Does the Activity Result in More Than a Minimal Increase in the Likelihood of Occurrence of a Malfunction of an SSC Important to Safety?.” The section states, in part: “although this criterion allows minimal increases, licensees must still meet applicable regulatory requirements and other acceptance criteria to which they are committed” (such as contained in regulatory guides and nationally recognized industry consensus standards, e.g., the ASME Boiler and Pressure Vessel Code and Institute of Electrical and Electronics Engineers standards). Further, departures from the design, fabrication, construction, testing and performance standards as outlined in the General Design Criteria (Appendix A to Part 50) are not compatible with a “no more than minimal increase standard.” In addition, changes in design requirements for earthquakes, tornadoes and other natural phenomena should be treated as potentially affecting the likelihood of malfunction. Based upon the above, the inspectors concluded that the change would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety, and therefore a license amendment was required prior to implementing the change.

The licensee documented the issue in AR 02534694. In response to this issue, the licensee has resumed entering applicable TSs anytime the inner railroad bay door is opened by entering the applicable TS action statements, evaluating weather conditions and impact to plant risk, and establishing the necessary mitigating actions required prior to opening the door.

Analysis. The inspectors determined that the licensee’s failure to provide a written evaluation describing the basis for determining that the change to the secondary containment completed on December 18, 2014, did not require a license amendment was a performance deficiency. Specifically, the licensee made a change to the secondary containment pursuant to 10 CFR 50.59(c) and eliminated the tornado wind and tornado missile loading condition from the FB Railroad Airlock (the enclosure walls and roof) and associated outer door (1SD1-31) Seismic Category I requirements, and did not provide a written evaluation describing the basis for determining that this change would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. The performance deficiency was determined to be more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," dated September 7, 2012, because it was associated with the design control attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers (fuel cladding, reactor coolant system and containment) protect the public from radionuclide releases caused by accidents or events. In addition, the associated violation was determined to be more than minor because the inspectors could not reasonably determine if the changes to secondary containment would have required NRC prior approval.

Violations of 10 CFR 50.59 are dispositioned using the traditional enforcement process instead of the significance determination process (SDP) because they are considered to be violations that potentially impede or impact the regulatory process. However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the inspectors used IMC 0609, Attachment 4, "Initial Characterization of Findings," and Appendix A, "The Significance Determination Process for Findings at Power," issued June 19, 2012, to evaluate the technical issue. The finding was screened against the Barrier Integrity cornerstone and determined to be of very low safety significance (Green) because the finding did not represent a degradation only of the radiological barrier function for the standby gas treatment system, nor did it represent a degradation of the function of the control room against smoke or toxic atmosphere.

The inspectors determined this finding had a cross-cutting aspect of procedure adherence in the area of human performance where individuals follow processes, procedures and work instructions. Specifically, the licensee failed to follow the 50.59 regulatory process as defined in station procedure LS-AA-104-1000, "50.59 Resource Manual," Revision 9. [H.8]

In accordance with Section 6.1.d.2 of the NRC Enforcement Policy, this violation was categorized as Severity Level IV because the resulting changes were evaluated by the SDP as having very low safety significance.

Enforcement. Title 10 CFR 50.59, "Changes, Tests and Experiments," states, in part, that a licensee shall maintain records of changes in the facility or procedures, and that the records must include a written evaluation that provides the bases for the determination that the change does not require a license amendment pursuant to 10 CFR 50.59(c)(2).

Contrary to the above, for a change to the secondary containment completed on December 18, 2014, the licensee did not provide a written evaluation describing the basis for determining that the change did not require a license amendment. Specifically, the licensee made a change to the secondary containment pursuant to 10 CFR 50.59(c) and eliminated the tornado wind and tornado missile loading condition from the FB railroad airlock (the enclosure walls and roof) and associated outer door

(1SD1-31) Seismic Category I requirements without providing a written evaluation describing the basis for determining that this change would not result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. In accordance with the Enforcement Policy, the violation was classified as a Severity Level IV violation because the underlying technical issue was of very low risk significance. Because this finding was of very low safety significance, was not repetitive or willful, and was entered into the licensee's CAP as AR 02534694, this violation is being treated as an NCV, in accordance with Section 2.3.2 of the NRC Enforcement Policy. The corrective action performed in response to this issue was to comply with

TS anytime the inner railroad bay door is opened by entering the applicable TS action statements, evaluating weather conditions and impact to plant risk and establishing the necessary mitigating actions required prior to opening the door. **(NCV 05000461/2015003-03, Failure to Obtain a License Amendment prior to Making Modifications to Secondary Containment.)**

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- Testing of the automatic depressurization system backup bottles and secondary filter;
- Testing of safety relief valves;
- Testing of fire pump B;
- Testing of RHR “C” room cooler valves; and
- Testing of 1SX063A, SX pump minimum flow DG 1A heat exchanger outlet valve.

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

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

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:

- CPS 9080.03, "Diesel Generator 1C Operability - Manual and Quick Start Operability," Revision 34d (Routine Test);
- CPS 9015.06, "Cold Shutdown Standby Liquid Control Pump and Valve Operability Check," Revision 29c (In-service Test); and
- CPS 9434.03, "ATWS Logic System Functional," Revision 34h (Routine Test).

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

- equipment was returned to a position or status required to support the performance of its safety functions; and
- all problems identified during the testing were appropriately documented and dispositioned in the CAP.

Documents reviewed are listed in the Attachment to this report.

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

b. Findings

No findings were identified.

Cornerstone: Emergency Preparedness

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a routine licensee emergency drill on September 29, 2015, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. The inspectors observed emergency response operations in the simulator and technical support center to determine whether the event classification, notifications, and protective action recommendations were performed in accordance with procedures. The inspectors also attended the licensee drill critique to compare any inspector-observed weakness with those identified by the licensee staff in order to evaluate the critique and to verify whether the licensee staff was properly identifying weaknesses and entering them into the CAP. The inspectors reviewed the drill package as well as other documents to ensure adequacy of drill control and assessment of the drill.

This emergency preparedness drill inspection constituted one sample as defined in IP 71114.06–05.

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Mitigating Systems Performance Index—Emergency Alternate Current Power System

a. Inspection Scope

The inspectors sampled licensee submittals for the Mitigating Systems Performance Index (MSPI) - Emergency Alternate Current (AC) Power System performance indicator (PI) for the period from the third quarter 2014 through the second quarter 2015. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, MSPI derivation reports, issue reports, event reports and NRC integrated inspection reports for the period of July 1, 2014 – June 30, 2015 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none was identified.

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

b. Findings

No findings were identified.

.2 Mitigating Systems Performance Index—Cooling Water Systems

a. Inspection Scope

The inspectors sampled licensee submittals for the MSPI - Cooling Water Systems PI for the period from the third quarter 2014 through the second quarter 2015. To determine the accuracy of the PI data reported during those periods, PI definitions and guidance contained in the NEI 99–02, “Regulatory Assessment Performance Indicator Guideline,” Revision 7, dated August 31, 2013, were used. The inspectors reviewed the licensee’s operator narrative logs, issue reports, MSPI derivation reports, event reports and NRC integrated inspection reports for the period of July 1, 2014 – June 30, 2015 to validate the accuracy of the submittals. The inspectors reviewed the MSPI component risk coefficient to determine if it had changed by more than 25 percent in value since the previous inspection, and if so, that the change was in accordance with applicable NEI guidance. The inspectors also reviewed the licensee’s issue report database to determine if any problems had been identified with the PI data collected or transmitted for this indicator and none was identified.

This inspection constituted one MSPI cooling water systems sample as defined in IP 71151–05.

b. Findings

No findings were identified.

4OA2 Identification and Resolution of Problems (71152)

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

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

a. Inspection Scope

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

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

b. Findings

No findings were identified.

.2 Daily Corrective Action Program Reviews

a. Inspection Scope

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

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

.1 (Closed) LER 05000461/2015-002-00, "Implementation of Enforcement Guidance Memorandum 11-003, Revision 2"

a. Inspection Scope

Between April 29, 2015, and May 12, 2015, CPS performed OPDRV activities while in Mode 5 without an operable secondary containment. An OPDRV is an activity that could result in the draining or siphoning of the reactor pressure vessel (RPV) water level below the top of the fuel, without crediting the use of mitigating measures to terminate the uncovering of fuel. Secondary containment is required by TS 3.6.4.1 to be operable during OPDRV. If secondary containment is inoperable during an OPDRV, TS requires the licensee to initiate actions to suspend the OPDRV immediately. Therefore, entering the OPDRV without establishing secondary containment integrity was considered a condition prohibited by TS.

As reported in Licensee Event Report (LER) 05000461/2015-002-00, CPS conducted the following OPDRVs during the period of secondary containment inoperability:

- Shifting RPV level control to RHR system;
- Nuclear instrumentation dry tube replacements;
- Control rod drive mechanism exchange;
- RHR shutdown cooling valves, 1E12F008 and 1E12F009, category A leak rate testing;
- Startup of reactor water cleanup (RWCU) system during system restoration;
- Startup of RHR A for shutdown cooling operations; and
- Align RWCU to reject to main condenser for RPV level control.

The NRC issued Enforcement Guidance Memorandum (EGM) 11-003, Revision 2, on December 13, 2013, to provide guidance on dispositioning boiling water reactor licensee non-compliance with TS containment requirements during OPDRV operations. The NRC considers enforcement discretion related to secondary containment inoperability during Mode 5, OPDRV activities to be appropriate as long as the licensee has taken the following interim actions: (1) adhere to the NRC plain language meaning of OPDRV activities, (2) meet the requirements which specify the minimum makeup flow rate and water inventory based on OPDRV activities with long drain down times, (3) ensure that adequate defense in depth is maintained to minimize the potential for the release of fission products with secondary containment not operable by (a) monitoring RPV level to identify the onset of a loss of inventory event, (b) maintaining the capability to isolate the potential leakage paths, (c) prohibiting Mode 4 (cold shutdown) OPDRV activities, and (d) prohibiting movement of irradiated fuel with the spent fuel storage pool gates removed in Mode 5, and (4) ensure that licensees follow all other Mode 5 TS requirements for OPDRV activities.

The inspectors reviewed this LER for potential performance deficiencies and/or violations of regulatory requirements. The inspectors also reviewed the stations implementation of the EGM during OPDRVs:

- The inspectors observed that, as required by the EGM, the OPDRV activities were logged in the control room narrative logs and that the log entry documented actions being taken to ensure water inventory was maintained and defense-in-depth criteria were in place.

- The inspectors noted that the reactor vessel water level was maintained at least 22 feet and 8 inches over the top of the RPV flange as required by TS 3.9.6. The inspectors also verified that at least one safety-related pump was the standby source of makeup designated in the control room narrative logs for the evolutions. The inspectors confirmed that the worst case estimated time to drain the reactor cavity to the RPV flange was greater than 24 hours.
- The inspectors reviewed engineering change documents which calculated the time to drain down during these activities and the feasibility of pre-planned actions the station would take to isolate potential leakage paths during these periods of time.
- The inspectors verified that the OPDRVs were not conducted in Mode 4 and that the licensee did not move irradiated fuel during the OPDRVs. The inspectors noted that CPS had a contingency plan in place for isolating the potential leakage path and verified that two independent means of measuring RPV water level were available for identifying the onset of loss of inventory events.
- The inspectors verified that, for the period of April 29 through May 2, and May 6 – 12, all other TSs were met during OPDRVs with secondary containment inoperable. For the period of May 3 – 6, inspectors identified that the requirements of TS 3.3.6.1 “Primary Containment and Drywell Isolation Instrumentation” and TS 3.3.6.2 “Secondary Containment Isolation Instrumentation,” were not met when the licensee failed to recognize the radiation monitor instrumentation was inoperable when the containment and drywell ventilation dampers were closed. Enforcement associated with this performance deficiency is discussed in the findings section below.

Technical Specification 3.6.4.1 requires, in part, that secondary containment shall be operable during OPDRVs. Technical Specification 3.6.4.1, Condition C, requires the licensee to initiate action to suspend OPDRV immediately when secondary containment is inoperable. Contrary to the above, between April 29, 2015 and May 2, 2015, and between May 6 – 12, 2015, CPS performed OPDRV activities while in Mode 5 without an operable secondary containment.

Because the violation occurred during the discretion period described in EGM 11-003, Revision 2, the NRC is exercising enforcement discretion in accordance with Section 3.5, “Violations Involving Special Circumstances,” of the NRC Enforcement Policy and, therefore, will not issue enforcement action for this violation.

In accordance with EGM 11-003, Revision 2, each licensee that receives discretion must submit a license amendment request within 4 months of the NRC staff’s publication in the Federal Register of the notice of availability for a generic change to the standard TS to provide more clarity to the term OPDRV. The inspectors observed that CPS is tracking the need to submit a license amendment request in its CAP as AR 1273398.

This LER (05000461/2015-002-00) is now closed. This inspection constituted one event follow-up sample as defined in IP 71153–05.

b. Findings

Failure to Enter Appropriate Technical Specification Action Statement for Inoperable Radiation Monitors during Operations with a Potential for Draining the Reactor Vessel Activities

Introduction. The inspectors identified a finding of very low safety significance, and an associated NCV of T.S. 3.3.6.1, “Primary Containment and Drywell Isolation Instrumentation,” and TS 3.3.6.2, “Secondary Containment Isolation Instrumentation,” for the licensee’s failure to enter the appropriate TSs and take the required actions associated with having inoperable containment radiation monitoring instrumentation during OPDRV activities. Specifically, with the containment ventilation dampers closed, the containment radiation monitoring instrumentation would not be able to perform its safety function of sending a containment isolation signal for elevated containment radiation levels as required during OPDRV activities.

Description. During the CPS C1R15 refueling and maintenance outage, the licensee performed maintenance on the instrument air (IA) system which impacted the containment building heating, ventilation and air conditioning system and the drywell purge systems. The ventilation dampers for these systems rely on IA to remain open; therefore, during the IA maintenance, the dampers were closed and the systems were isolated. The containment ventilation ducts support the radiation monitoring function for monitoring elevated radiation levels in containment during certain modes of applicability including OPDRVs. Specifically, the containment building fuel transfer pool ventilation plenum exhaust radiation monitor, the containment building exhaust radiation monitor and the containment building continuous containment purge exhaust radiation monitor take measurements from the ventilation ducts. Based on the radiation levels measured in these ducts, the instrumentation associated with these monitors will send a signal to a specific set of components to isolate primary and secondary containment. The primary purpose of these instruments is to detect gross failure of the fuel cladding and to ensure offsite doses remain below 10 CFR Part 20 and 10 CFR Part 100 limits.

Technical Specification 3.3.6.1 states, “The primary containment and drywell isolation instrumentation for each function in Table 3.3.6.1-1 shall be operable. In addition, T.S. 3.3.6.2 states, “The secondary containment isolation instrumentation for each function in Table 3.3.6.2-1 shall be operable. Both Table 3.3.6.1-1 and Table 3.3.6.2-1 state the containment building fuel transfer ventilation plenum-high, the containment building exhaust radiation high and the containment building continuous containment purge exhaust radiation-high functions shall be operable during movement of recently irradiated fuel assemblies as well as during OPDRVs.

For T.S. 3.3.6.1, Condition D, “One or more required channels inoperable,” the required action is to place the channel in trip in 24 Hours. For Condition E, “One or more automatic functions with isolation capability not maintained,” the required action is to restore isolation capability within 1 hour. If neither of these actions could be met, the required action is to initiate action to suspend the OPDRV or isolate the affected penetrations immediately. For the same conditions as those stated above, T.S. 3.3.6.2 requires the isolation of associated penetration flow paths or declaring the associated containment isolation dampers inoperable and placing the associated standby gas treatment system in operations or declaring the associated standby gas treatment system inoperable.

Per TS, a system, subsystem, division, component or device shall be operable or have operability when it is capable of performing its specified safety function and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication and other auxiliary equipment that are required for the system, subsystem, division, component or device to perform its specified safety functions are also capable of performing their related support function.

From 2132 on May 2, 2015, through 0940 on May 12, 2015, the licensee took the containment building heating, ventilation and air conditioning system and the drywell purge systems out of service for planned maintenance. While the two systems were out of service, the aforementioned radiation monitors were isolated from the containment atmosphere because there was no flow through the ventilation ducts with the dampers closed. Therefore, the monitors would not be able to fulfill their safety function of sending an isolation signal to various components in order to isolate primary and secondary containment in the case of elevated radiation levels in containment during any OPDRV that was in progress.

The licensee did not recognize that the radiation monitors were inoperable during this timeframe. The licensee based their operability determination on the radiation monitor instrumentation being able to pass its surveillance tests, which consisted of channel checks, channel calibrations and logic functional tests. They did not take into consideration the impact of the ventilation dampers being closed on the monitor's ability to detect elevated radiation levels in containment.

From May 3, 2015, through May 6, 2015, the licensee executed two OPDRV windows in excess of 1 hour. During this timeframe, the affected penetrations were not isolated, standby gas treatment was not in service and OPDRVs were not suspended immediately. Therefore, the inspectors determined that the licensee did not comply with T.S. 3.3.6.1 and TS 3.3.6.2 action statements.

The NRC issued EGM 11-003, Revision 2, "Dispositioning BWR Licensee Non-Compliance with TS Containment Requirements during Operations with a Potential for Draining the Reactor Vessel (OPDRVs)," to exercise enforcement discretion and not cite licensees for TS violations related to conduct of OPDRVs with secondary containment inoperable provided that certain criteria were met. One of those criteria was that the licensee must follow all other TS applicability and action requirements for Mode 5. Since CPS was conducting OPDRVs during the time of the radiation monitor instrumentation inoperability, CPS did not meet the criteria in EGM 11-003 for the staff to consider exercising discretion from May 3 – 6, 2015. For the radiation monitor inoperability, T.S. 3.3.6.1 and TS 3.3.6.2 required initiation of actions to suspend OPDRVs immediately, as did TS 3.6.4.1, for inoperable secondary containment. Therefore, the licensee was in a condition prohibited by TS.

<u>Analysis:</u> The inspectors determined that the failure to enter T.S. 3.3.6.1 and TS 3.3.6.2 when the radiation monitor instrumentation was not able to perform its safety function during an OPDRV was a performance deficiency. Specifically, the licensee failed to recognize that when the containment ventilation dampers were closed, the radiation monitors could not detect the radiation levels in primary containment and therefore could not fulfill their safety function of sending containment isolation signals in the case of elevated radiation levels in containment. The performance deficiency was more than minor in accordance with IMC 0612, "Power Inspection Reports," Appendix B, "Issue

Screening," dated September 7, 2012, because it was associated with the SSC and Barrier Performance attribute of the Barrier Integrity cornerstone, and adversely affected the cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events, and is therefore a finding. Specifically, the automatic containment isolation signal function of the radiation monitors was impacted when the containment ventilation dampers were closed during OPDRV activities.

Using IMC 0609, Attachment 4, "Initial Characterization of Findings," dated June 19, 2012, and Appendix G, Attachment 1, "Shutdown Operations Significance Determination Process Phase 1 Initial Screening and Characterization of Findings," dated May 9, 2014, the finding was screened against the Barrier Integrity cornerstone, and determined to need a detailed risk evaluation because the finding represents a degradation of the ability to close or isolate the containment.

Using Appendix G Exhibit 4, "Barrier Integrity Screening Questions," the Region III senior reactor analyst (SRA) determined that the finding degraded the ability to close or isolate the containment per Section B, "Containment Barrier," Question 6. Therefore, the evaluation was continued using IMC 0609 Appendix H, "Containment Integrity Significance Determination Process." The SRA determined this to be a "Type B" finding because it was related to a degraded condition that had implications for containment integrity without affecting the likelihood of core damage. The SRA used Section 6.2 of Appendix H, "Approach for Assessing Type B Findings at Shutdown." Based on information from the inspectors, during all OPDRV time windows, the reactor water level was confirmed to be greater than the minimum level required for movement of irradiated fuel assemblies (i.e., greater than 22'8" above the flange). This plant condition meets the definition of "Plant Operating State 3 of Appendix H. Therefore, since the plant was in Plant Operating State 3 during the OPDRV time windows, the finding screens as of very low safety significance (Green) per Step 2.1 of Section 6.2 of Appendix H.

The inspectors determined this finding had a cross-cutting aspect of conservative bias in the area of human performance where individuals use decision making practices that emphasize prudent choices over those that are simple allowable. A proposed action is determined to be safe in order to proceed, rather than unsafe in order to stop. Specifically, the licensee relied solely on the successful completion of the surveillance requirements to determine the radiation monitor instrumentation was operable rather than considering the impact the closed dampers would have on their ability to fulfill their safety function. [H.14]

Enforcement: TS 3.3.6.1 states, "The primary containment and drywell isolation instrumentation for each function in Table 3.3.6.1-1 shall be operable." TS 3.3.6.2 states, "The secondary containment isolation instrumentation for each function in Table 3.3.6.2-1 shall be operable." Both Table 3.3.6.1-1 and Table 3.3.6.2-1 stated the containment building fuel transfer ventilation plenum-high, the containment building exhaust radiation-high and the containment building continuous containment purge exhaust radiation-high functions shall be operable during movement of recently irradiated fuel assemblies as well as during OPDRV activities.

Contrary to the above, from May 3, 2015 through May 6, 2015, the licensee failed to ensure the containment building fuel transfer ventilation plenum-high, the containment building exhaust radiation-high and the containment building continuous containment purge exhaust radiation-high functions were operable during OPDRVs. Specifically, with the containment building heating, ventilation, air conditioning system and the drywell purge system ventilation dampers closed, the instrumentation would not be capable of detection radiation levels in the containment environment and therefore would not be able to fulfill its safety function of sending isolation signals to various components on a high radiation signal. Consequently, since the criteria for exercising enforcement discretion, in accordance with EGM 11-003, were not met during this time frame, this condition also represents a violation of TS 3.6.4.1 for inoperable secondary containment during OPDRVs. This issue was identified subsequent to the licensee exiting the IA maintenance as well as the OPDRV activities, therefore the TS noncompliance was no longer in effect. Because this violation is of very low safety significance and was entered into the licensee's CAP as AR 02566708, this violation is being treated as a NCV consistent with Section 2.3.2 of the NRC Enforcement Policy. **(NCV 05000461/ 2015003-04, Failure to enter appropriate TS action statement for inoperable radiation monitors during OPDRV activities)**

4OA6 Management Meetings

.1 Exit Meeting Summary

On October 15, 2015, the inspectors presented the inspection results to Mr. D. Kemper 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.

.2 Interim Exit Meetings

An interim exit was conducted for:

- The results of the in-service inspection with Mr. T. Stoner, Plant Manager and other members of the licensee staff on August 14, 2015.

The inspectors confirmed that none of the potential report input discussed was considered proprietary. Proprietary material received during the inspection was returned to the licensee.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Bair, Chemistry Manager
M. Friedmann, Emergency Preparedness Manager
B. Brooks, Security Manager
J. Cunningham, Maintenance Director
C. Dunn, Training Director
N. Hightower, Radiation Protection Manager
T. Krawcyk, Shift Operations Superintendent
D. Kemper, Acting Plant Manager/Operations Director
J. Blount, Acting Senior Manager Design Engineering
M. Newcomer, Site Vice President
C. Propst, Work Management Director
J. Ward, Nuclear Oversight Manager
D. Shelton, Operations Services Manager
J. Smith, Engineering Director
S. Minya, Operations Training Manager
T. Stoner, Plant Manager
M. Heger, Acting Senior Manager Plant Engineering
G. Engelhardt, Engineering Programs Manager
D. Anthony, Exelon NDE Services
M. Baig, ISI Program Engineer

U.S. Nuclear Regulatory Commision

K. Stoedter, Chief, Reactor Projects Branch 1
W. Schaup, Clinton Senior Resident Inspector
E. Sanchez-Santiago, Clinton Resident Inspector

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000461/2015003-01	NCV	Failure to Follow Procedure leaves Control Room Cabinet doors Unattended in Seismically Unanalyzed Condition (Section 1R15)
05000461/2015003-02	NCV	Failure to Implement and Comply with Transient Equipment/Materials Program (Section 1R15)
05000461/2015003-03	NCV	Failure to Obtain a License Amendment prior to Making Modifications to Secondary Containment (Section 1R18)
05000461/2015003-04	NCV	Failure to enter appropriate TS action statement for inoperable radiation monitors during OPDRV activities (Section 4OA3)

Closed

05000461/2015002-01	URI	Fuel Handling Building Railroad Bay Airlock Design and Licensing Basis Issues (Section 1R18)
05000461/2015-002-00	LER	Implementation of Enforcement Guidance Memorandum (EGM) 11-003, Revision 2 (Section 4OA3)

LIST OF DOCUMENTS REVIEWED

1R04 Equipment Alignment

- UFSAR Section 4.6, “Design of Reactivity Control Systems,”
- CPS 3304.01V001, “Control Rod Hydraulic and Control Valve Lineup”; Rev. 24b
- CPS 3304.01, “Control Rod Hydraulic & Control”; Rev. 35d
- CPS 3304.01E001, “Control Rod Drive Hydraulic Electrical Lineup”; Rev. 7
- CPS 3304.01E002, “CRD 120 VAC Electrical Lineup”; Rev. 4b
- CPS 5068.05, “Alarm Palen 5068 Annunciators – Row 5”; Rev. 24b
- CPS 3304.01V002, “CRD Instrument Valve Lineup”; Rev. 12
- CPS 3506.01V001; Diesel Generator and Support Systems Valve Lineup; Rev. 12a
- CPS 3506.01V002; Diesel Generator and Support Systems Instrument Valve Lineup; Rev. 11b
- CPS 3506.01E001; Diesel Generator and Support Systems Electrical Lineup; Rev. 18c
- CPS 3506.01; Diesel Generator and Support Systems; Rev. 37a
- CPS 3312.01V002, “ RHR System C Instrumentation Valve Lineup”; Rev. 9a
- CPS 3312.01E001, “Residual Heat Removal Electrical Lineup”; Rev. 17
- CPS 3312.01V001, “RHR Valve Lineup”; Rev. 17b
- CPS 3317.01E001, “Fuel Pool Cooling and Cleanup Electrical Lineup”; Rev. 13b
- CPS 3317.01V002, “FPCC Instrument Valve Lineup”; Rev. 7
- CPS 3317.01V001, “Fuel Pool Cooling and Cleanup Valve Lineup”; Rev. 12a
- M05-1078, “Control Rod Drive P&ID,” Rev. J
- AR 02505401, “CRD Pump Suction Pressure Oscillations”
- AR 02543383, “NRC Asked Question About White Crystal Residue On HCU 32-25,” UFSAR Section 4.6, “Design of Reactivity Control Systems,”
- AR 02503386, “1PC-CY007 Loop Causes CRD Inlet Pressure Oscillations”
- AR 02543321, “1C11D001BQ: Scram Pilot Solenoid Valve F139 – Small Air Leak”

1R05 Fire Protection

- Clinton Power Station Updated Final Safety Analysis Report, Appendix E, “Fire Protection Evaluation Report – Clinton Power Station Unit 1”; Rev. 17
- CPS 1019.05, “Transient Equipment/Materials”; Rev. 23a
- CPS 1893.01, “Fire Protection impairment Reporting”; Rev. 20d
- CPS 3213.01, “Fire Detection and Protection”; Rev. 29d
- OP-AA-201-004, “Fire Prevention for Hot Work”; Rev. 12
- OP-AA-201-008, “Prefire Plan Manual”; Rev. 3
- OP-AA-201-009, “Control of Transient Combustible Material”; Rev. 13
- OP-MW-201-007, “Fire Protection System Impairment Program”; Rev. 7
- CPS 1893.04M511, “Division 1 Diesel Generator & Day Tank Room Prefire Plan”; Rev. 6a
- CPS 1893.04M501, “Division 1 Diesel Fuel Tank Room Prefire Plan”; Rev. 5
- CPS 1893.04M431, “781 Fuel: East Balcony Prefire Plan”; Rev. 2
- CPS 1893.04M740, “800 Turbine: Turbine Deck Prefire Plan”; Rev. 5a
- CPS 1893.04M400, “712 Fuel: Basement Prefire Plan”; Rev. 5
- CPS 1983.04M410, “737 Fuel: Grade Level Prefire Plan”; Rev. 4b
- CPS 1893.04M104, “707 Auxiliary: RHR B Pump and Heat Exchanger Room Prefire Plan”; Rev. 5
- CPS 1893.04M420, 755 Fuel: Fuel Handling Floor Prefire Plan”; Rev. 4

1R06 Flooding Protection Measures

- CPS 4304.01, “Flooding”; Rev. 6
- Analysis 3C10-0485-001, Internal Flood Analysis; Rev. 8-E
- Analysis 3C10-0485-001, Internal Flood Analysis; Rev. 9
- WO 01754104, Verify Functionality of Seven Important Floor Drains; dated 01/15/15
- M01-1600, Environmental Zone Map, Control Building Floor Plan El. 781’-0”; Rev. A
- WO 01799135, Verify Functionality of Seven Important Floor Drains; dated 07/15/15
- AR 2542996, No Flow Alarm on 1FP15SA During Yearly Surveillance Again; dated 08/18/15
- CY-CL-3221-02, “Operating Cable Vault Pumping Stations”; Rev. 6

1R07 Heat Sink Performance

- CPS 8801.12C001, “Local Mounted Instrument Valve Operation Checklist”; Rev. 15
- CPS 2700.14C001, “SX Flow Verification Test Instrumentation – Division III”
- CPS 2700.14D001, “SX Flow Verification Data Sheet – Division III”; dated 12/11/14
- WO 1628381, “Perform Division III SX System Testing IAW 2700.14”
- AR 2528112, “1VH07SC Not In Scope During Division III SX Flow Balance”; dated 07/14/15
- AR 2528036, “Received Ann. 5064-2A Not Available Division 3 SX”; dated 07/14/15

1R08 Inservice Inspection Activities

- Work Order 01356031-02, Install Blind Coupling Per EC 371540, VT-2; dated 05/03/12
- Work Order 01356031-01, Install Blind Coupling Per EC 371540; dated 05/03/12
- Report RF-14-008, NIS-2 Form for Blind Coupling Replacement Per EC 371540; dated 08/14/12
- AR 0249886, INR C1R15 IVVI-15-03, Steam Dryer OD Access Hole Patch 325 D; dated 05/0315
- Procedure CPS 9843.02, Operational Pressure Testing of Class 1, 2, and 3 Systems, Rev. 43a
- Procedure CPS 9861.02, Local Leak Rate Testing Requirements and Type C (Air) Local Leak Rate Testing; Rev. 44
- Report C1R15-ISI-008, VT-3 Visual Examination NDE Report for Component Supports, Attachments and Interiors of Reactor Vessels, 1RH07074X; dated 04/28/15
- Procedure CPS 1305.01, Primary Containment Leakage Rate Testing Program; Rev. 12
- Procedure CPS 9861.01, Integrated Leak Rate Test; Rev. 26a
- Report C1R15-ISI-004, VT-3 Visual Examination NDE Report for Component Supports, Attachments and Interiors of Reactor Vessels, 1SX01004R; dated 04/27/15
- Report C1R15-ISI-006, VT-3 Visual Examination NDE Report for Component Supports, Attachments and Interiors of Reactor Vessels, 1SX22003A; dated 04/27/15
- Engineering Change Package (EC) 371540, Install Blind Coupling on line 1SC27A near Penetration 1MCC116 and Abandoned in Place Isolation Valves 1C41F340B/F341B; Rev. 1
- Work Order 01312496-01, Replace HPCS Water Leg Pump Discharge Stop Check Valve; dated 10/19/13
- AR 01576806, C1R14 – FME, Dropped Nut Between RPV and Insulation Pack; dated 10/25/13
- AR 01571950, FME in SW Corner of Dryer Pool - 2 Inch Piece of Duct Tape; dated 10/14/13
- AR 01571318, C1R14 – Historical FME Identified In Reactor Vessel; dated 10/12/13
- AR 02496621, NRC ISI Inspection Concern; dated 05/06/15
- AR 02496617, NRC ISI Inspection Concern; dated 05/06/15

1R11 Licensed Operator Requalification Program

- TQ-AA-155, “Conduct of Simulator Training and Evaluation”; Rev. 5
- EP-AA-125-1002, “Emergency Response Organization Performance Indicators Guidance”; Rev. 9
- OP-AA-101-111-1001, “Operations Standards and Expectations”; Rev. 17
- OP-CL-108-101-1003, “Operations Department Standards and Expectation”; Rev. 35
- TQ-AA-150, “Operator Training Programs”; Rev. 12
- CPS 9031.06, “Main Turbine Stop Valve and Combined Intermediate Valve Tests”; Rev. 34c
- CPS 9031.07, “Main Turbine Control Valve Tests”; Rev. 33d
- CPS 9031.10, “RPS Main Steam Line Isolation Valve Channel Functional”; Rev. 25d
- CPS 3005.01, “Unit Power Changes”; Rev.42d
- CPS 3005.01F001, “Unit Power Changes Power Increase Flow Chart”; Rev. 0
- CPS 3005.01F002, “Unit Power Changes Power Decrease Flow Chart”; Rev. 0
- SE-LOR-445 (Simulator Scenario); Rev. 0

1R12 Maintenance Effectiveness

- ER-AA-310, “Implementation of Maintenance Rule”; Rev. 9
- ER-AA-310-1001, “Maintenance Rule Scoping”; Rev. 4
- ER-AA-310-1002, “Maintenance Rule Functions – Safety Significance Classification”; Rev. 3
- ER-AA-310-1003, “Maintenance Rule – Performance Criteria Selection”; Rev. 4
- ER-AA-310-1004, “Maintenance Rule – Performance Monitoring”; Rev. 13
- ER-AA-310-1005, “Maintenance Rule – Dispositioning Between (a)(1 and (a)(2)”; Rev. 7
- ER-AA-310-1006, “Maintenance Rule – Expert Panel Roles and Responsibilities”; Rev. 5
- ER-AA-310-1007, “Maintenance Rule – Periodic (a)(3) Assessment”; Rev. 4
- NUMARC 93-01, “Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants”; Rev. 2
- 10 CFR 50.65 (a)(3) Periodic Assessment of Maintenance Rule Program Clinton Power Station March 1, 2014 to May 17, 2015
- AR 02529702, “Issues Identified During MRule Assessment”

1R13 Maintenance Risk Assessments and Emergent Work Control

- AD-AA-3000, “Nuclear Risk Management Process”; Rev. 1
- ER-AA-600, “Risk Management”; Rev. 7
- ER-AA-600-1011, “Risk Management Program”; Rev. 14
- ER-AA-600-1012, “Risk Management Documentation”; Rev. 12
- ER-AA-600-1014, “Risk Management Configuration Control”; Rev. 7
- ER-AA-600-1042, “On-line Risk Management”; Rev. 9
- OP-AA-108-117, “Protected Equipment Program”; Rev. 4
- WC-AA-101, “On-Line Work Control Process”; Rev. 25
- WC-AA-104, “Integrated Risk Management”; Rev. 23

1R15 Operability Evaluations

- CC-AA-309-101, “Engineering Technical Evaluations”; Rev. 14
- OP-AA-108-104, “Technical Specification Compliance”; Rev. 1
- OP-AA-108-115, “Operability Determinations (CM-1) ”; Rev. 16
- OP-AA-108-115-1002, “Supplemental Consideration for On-shift Immediate Operability Determinations (CM-1) ”; Rev. 3
- CPS 1014.11, “6900/4160/480V Switchgear/Circuit Breaker Operability Program”; Rev. 5a

- CPS 1014.11, "6900/4160/480V Switchgear/Circuit Breaker Operability Program"; Rev. 5b
- CPS 1019.05, "Transient Equipment/Materials"; Rev. 23
- Drawing E03-1RF00, Sheet 2; Rev. J
- Drawing E02-1RF99, Sheet 4; Rev. N
- Drawing E02-1RF99, Sheet 8; Rev. N
- Drawing E02-1RF99, Sheet 10; Rev. P
- EC 397681, Past Operability Technical Assessment for IR 1630607"; Rev. 0
- AR 02541236, "Part 21: NAMCO Limit Switch (EA170/EA180)"
- AR 02494259, "NRC SRI Question Concerning MCR Panel Doors"
- AR 02512414, "NRC Observation of MCR Back Panel Door Control"
- AR 02518477, "NRC Observation of MCR Back Panel Door Control"
- AR 02523496, "0FP01PB Data From 9071.02 Requires Evaluation"
- AR 02529227, "NRC Questions Seismic Qualifications of Installed Tele-tower"
- AR 02507167, "Noncompliance with CPS1019.05"
- AR 01630607, "IEMA Obs/Questions about 707 Ab Storage Area and Vent Panels"
- AR 02550937, "Past Operability Not Performed for Materials Near PL62J"
- AR 02547146, "MOS ID: 1019.05 Approved Storage Area Requirement Not Met"
- WO 01849368, "5009-3A Activated Seismic Recorder"; dated 07/29/15

<u>1R18 Plant Modifications</u>

- EC 377321, "Isolate Floor Drain Line LPCS Pump Room and Revise the Flood Plan for RHR-A Pump Room and the Radwaste Pipe Tunnel"; Rev. 0
- WO 01274884, "Blank Drain Line at V-124 707 El-LPCS Room"; dated 10/07/09

<u>1R19 Post-Maintenance Testing</u>

- MA-AA-716-012, "Post-Maintenance Testing"; Rev. 20
- CPS 9071.02, "Diesel Fire Pump Capacity Checks"; Rev. 40c
- CPS 9071.02D001, "Diesel Fire Pump Capacity Check Data Sheet"; Rev. 29
- CPS 9071.01, "Diesel Driven Fire Pumps Operability Test"; Rev. 40
- CPS 9377.01, "Fire Protection Diesel 24VDC Battery Pilot Cell Check"; Rev. 31
- CPS 8377.01, "Fire Protection Diesel 24VDC Battery Maintenance"; Rev. 11
- CPS 8377.01C001, "24VDC Battery Maintenance Checklist"; Rev. 7
- CPS 8377.01F001, "24VDC Cattery Cable/Component Removal/Installation Form"; Rev. 4a
- CPS 8801.12C001, "Local Mounted instrument Valve Operation Checklist"; Rev. 15b
- CPS 9377.02, "Fire Protection Diesel 24VDC Battery Check"; Rev. 32
- CPS 9069.01, "Shutdown Service Water operability Test"; Rev. 48d
- CPS 9069.01D001, "Shutdown Service Water System Operability Data Sheet"; Rev. 46d
- CPS 9059.01, Reactor Coolant System Leakage Test"; Rev. 10
- CPS 9059.01V001, "Reactor Coolant System Leakage Test Valve Lineup"; Rev. 3
- AR 02523496, "0FP01PB Data From 9071.02 Requires Evaluation"
- AR 0250503, "1B33F357B Packing Leak 5dpm"
- AR 02535401, "NRC Question Regarding PMT Acceptance Criteria"
- AR 02500496, "1B33F354B Packing Leak on High Side Line Main Root"
- AR 02500492, "Leak Detected During RPV Pressure Test 1B21F040"
- AR 02500475, "Packing Leak on E22F005 PSU"
- AR 02500472, "Packing Leak on 1B21F022D PSU"
- WO 01781766, "0FP01PB Fire Pump Crankcase Pressure Check"
- WO 01759788, "Fire Pump B Capacity Test"
- WO 01840330, "Fire Pump Operability"

- WO 01770266, “2FP013 Fire Pump Relief Valve Inspection”
- WO 01752805, “Perform Maintenance Checks on 0FP01PB”
- WO 01697257, “Replace Batteries in 0FP01PB”
- WO 01697500, “Perform Trip-Point Calibration”
- WO 01836886, “Fire Protection Battery Pilot”
- WO 01842909, “SX Pump operability Test (SX pump A)”
- WO 01496131, “Replace Filter Cartridges”
- WO 01692556, “9059.01R20 LRT Vessel Pressure Test”; dated 05/12/15

1R22 Surveillance Testing

- CPS 9434.03, “ATWS Logic System Functional,” Revision 34h
- CPS 9080.03, “Diesel Generator 1C Operability – Manual and Quick Start Operability,” Revision 34d
- CPS 9080.03D001, “Diesel Generator 1C Operability – Manual and Quick Start Data Sheet,” Revision 23
- CPS 3506.01D003, “Diesel Generator Operating Logs,” Revision 6b
- CPS 3506.01C005, “Diesel Generator Start Log,” Revision 1b
- CPS 9015.06, “Cold Shutdown Standby Liquid Control Pump and Valve Operability Check,” Revision 29c
- CPS 9015.06D001, “Cold Shutdown Standby Liquid Control Pump and Valve Data Sheet,” Revision 29
- CPS 9843.02D001, “Generic Class 1, 2, and 3 Operational Pressure Test Data Sheet,” Revision 43
- AR 02499823, “Division 2 ATWS Time Delay Relay OOS”
- AR 02499305, “1C11-F-405A Failed to Stroke During ATWS LSF”
- AR 02545300, “Evaluate Work Impact with No Parts Identification”
- AR 02527149, “SLC Pump A Crankcase Oil Level 1/8” High (1C41C001A)”
- WO 01695384, “ATWS Logic System Functional (Division 2)”
- WO 01695383, “ATWS Logic System Functional (Division 1)”
- WO 01830744, “1C11-F-405A Failed to Stroke During ATWS LSF”
- WO 01687921, “SLC Pump A Operability (Refuel Outage for 1C41F033B)”
- WO 01843129, “DG 1C Operability – Monthly Test”

1EP6 Drill Evaluation

- EP-AA-125-1001, “Emergency Planning Performance Indicators Guidance”; Rev. 8
- EP-AA-125-1002, “Emergency Response Organization Performance Indicators Guidance”; Rev. 9
- EP-AA-125-1003, “Emergency Response Organization Readiness Performance Indicators Guidance”; Rev. 10
- EP-AA-125-1004, “Emergency Response Facilities and Equipment Performance Indicators Guidance”; Rev. 9
- EP-AA-125, “Emergency Planning Self Evaluation Process”; Rev. 9
- EP-AA-122, “Drills and Exercise Program”; Rev. 18
- EP-AA-122-200, “Drills and Exercise Execution”; Rev. 2
- EP-AA-122-300, “Drills and Exercise Evaluation”; Rev. 2

4OA1 Performance Indicator Verification

- MSPI Derivation Report, MSPI Emergency AC Power System Unavailability Index
- MSPI Derivation Report, MSPI Emergency AC Power System Unreliability Index

- MSPI Derivation Report, MSPI Cooling Water System Unavailability Index
- MSPI Derivation Report, MSPI Cooling Water System Unreliability Index

4OA2 Identification and Resolution of Problems

- CC-AA-10, “Configuration Control Process Description”; Rev. 8
- CC-AA-102, “Design Input and Configuration Change Impact Screening”; Rev. 28
- CC-AA-106-1001, “Configuration Change Walkdowns”; Rev. 5
- CC-AA-112, “Temporary Configuration Changes”; Rev. 22
- OP-AA-106-101-1006, “Operational Decision making Process”; Rev. 16
- PI-AA-125, “Corrective Action Program CAP Procedure”; Rev. 2
- PI-AA-125-1001, “Root Cause Analysis Manual”; Rev. 1
- PI-AA-125-1003, “Apparent Cause Evaluation Manual”; Rev. 2
- PI-AA-125-1004, “Effectiveness Review Manual”; Rev. 0
- PI-AA-125-1006, “Investigation Techniques Manual”; Rev. 1
- CPS 4301.01, “Earthquake”; Rev. 15a
- CPS 3323.01, “Seismic and Environmental Monitoring”; Rev. 11c
- CPS 9037.03, “ Triaxial Seismic Switch 1VS-EM014 Channel Functional”; Rev. 27a
- CPS 9037.21, “Triax Time-History Accelerometer Channel Functional”; Rev. 29b
- WO 1849368, “5009-3A Activated Seismic Recorder”; dated 07/29/15
- AR 02546209, “Triaxial Seismic Accekerometer 1VTEM002 Failure and Replace”

4OA3 Follow-up of Events and Notices of Enforcement Discretion

- CPS 4001.02, “Automatic Isolation”; Rev. 17c
- CPS 4001.02C001, “Automatic Isolation Checklist”; Rev. 16
- CPS 3007.01C005, “Operations with a Potential for Draining the Reactor Vessel Checklist”; Rev. 2c
- CPS 3007.01, “Preparation and Recovery form Refueling Operations”; Rev. 19d
- CPS 3312.03, “RHR Shutdown Cooling & Fuel Pool Cooling and Assist”; Rev. 10b
- AR 02494981, “Utilization of EGM 11-003 for OPDRVs”
- AR 02566708, “NRC Position on Containment Ventilation Radiation Monitors”

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Document Access Management System
AR	Action Request
ASME	American Society of Mechanical Engineers
CAP	Corrective Action Program
CFR	Code of Federal Regulation
CPS	Clinton Power Station
DG	Diesel Generator
EC	Engineering Change
EGM	Enforcement Guide Memorandum
FB	Fuel Building
IA	Instrument Air
IMC	Inspection Manual Chapter
IP	Inspection Procedure
ISI	Inservice Inspection
MSIV	Main Steam Isolation Valve
MSPI	Mitigating Systems Performance Index
NCV	Non-Cited Violation
NDEs	Non-Destructive Examinations
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OPDRV	Operations with a Potential for Draining the Reactor Vessel
PI	Performance Indicator
PARS	Publicly Available Records System
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SDP	Significance Determination Process
SRA	Senior Reactor Analyst
SSC	Systems, Structures, and Components
SX	Service Water
TS	Technical Specification
USAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VT	Visual Testing

Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Clinton Power Station. In addition, if you disagree with the cross-cutting aspect assigned to any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region III, and the NRC Resident Inspector at the Clinton Power Station.

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, its enclosure, and your response (if any) will be available electronically for public inspection in the NRC’s Public Document Room or from the Publicly Available Records (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/

Patrick Louden, Director
Division of Reactor Projects

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