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

August 3, 2010

Mr. Michael J. Pacilio
Senior Vice President, Exelon Generation Company, LLC
President and Chief Nuclear Officer (CNO), Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION NRC INTEGRATED INSPECTION REPORT 05000461/2010-003

Dear Mr. Pacilio:

On June 30, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at your Clinton Power Station. The enclosed report documents the inspection results, which were discussed on July 8, 2010, with Mr. M. Kanavos and other members of your staff.

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

Based on the results of this inspection, one self-revealed and four NRC-identified findings of very low safety significance were identified. Four of these findings were determined to involve violations of NRC requirements. Additionally, five licensee-identified violations, which were determined to be of very low safety significance, were reviewed by the inspectors and are listed in this report. Because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating the above violations as non-cited violations (NCVs) consistent with Section VI.A.1 of the NRC Enforcement Policy.

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 a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at 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 Clinton Power Station.

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

Sincerely,

/RA/

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

Docket No. 50-461
License No. NPF-62

Enclosure: Inspection Report 05000461/2010-003
w/Attachment: Supplemental Information

cc: Distribution via ListServe

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	05000461/2010-003
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station, Unit 1
Location:	Clinton, IL
Dates:	April 1 through June 30, 2010
Inspectors:	B. Kemker, Senior Resident Inspector D. Lords, Resident Inspector J. Cassidy, Senior Health Physicist E. Coffman, Reactor Engineer D. Meléndez-Colón, Resident Inspector, Dresden M. Mitchell, Health Physicist R. Russell, Emergency Preparedness Inspector S. Mischke, Resident Inspector, Illinois Emergency Management Agency
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000461/2010-003, 04/01/10 – 06/30/10; Clinton Power Station, Unit 1; Flood Protection Measures, Maintenance Risk Assessments and Emergent Work Control, Operability Evaluations, Emergency Response Organization Augmentation Testing, and Identification and Resolution of Problems.

This report covers a three-month period of inspection by the resident inspectors and announced baseline inspections by regional inspectors. Two Severity Level IV non-cited violations (NCVs) and three Green findings, two of which had an associated NCV, were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (SDP). Findings for which the SDP does not apply may be Green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Severity Level IV. The inspectors identified a finding of very low safety significance with an associated Severity Level IV non-cited violation of the NRC's reporting requirements in 10 CFR 50.72(a)(1), "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 10 CFR 50.73(a)(1), "Licensee Event Report System." The licensee failed make a required 8-hour non-emergency notification call to the NRC Operations Center and failed to submit a required Licensee Event Report within 60 days after discovery on October 7, 2009, of a condition that resulted in the plant being in an unanalyzed condition that significantly degraded plant safety and could have prevented fulfillment of the safety function of the emergency core cooling system. No immediate corrective actions were taken to address this finding; however, the licensee entered this issue into its corrective action program for evaluation.

  This finding was of more than minor significance because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the Technical Specifications and the regulations in order to perform its regulatory function. Because this issue affected the NRC's ability to perform its regulatory function, the inspectors evaluated it using the traditional enforcement process and assessed the significance of the underlying issue using the SDP. The underlying technical issue (i.e., interconnecting floor drains between the Residual Heat Removal 'A' Pump Room and the Radwaste Pipe Tunnel) was determined to be a finding of very low safety significance during a Phase 3 Significance Determination Process evaluation. Consistent with the guidance in Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the violation associated with this finding was determined to be a Severity Level IV Violation. This finding affected the cross-cutting area of human performance because the licensee did not use conservative assumptions in decision making while evaluating the reportability of the unanalyzed condition with respect to the reporting requirements in 10 CFR 50.72(a)(1)(ii) and 50.73(a)(1). (IMC 0310 H.1(b)) (Section 1R06.1.b.(1))

- Green. The inspectors identified a finding of very low safety significance associated with the licensee's failure to evaluate the functionality of multiple excess flow check valves that had not been tested in accordance with the American Society of Mechanical Engineers / American National Standards Institute (ASME/ANSI) Code Inservice Testing requirements to establish whether the nonconforming condition warranted starting the Technical Specification (TS) action time for the suppression pool makeup (SPMU) system. In response to the inspectors' questions, the licensee subsequently performed an operability evaluation. No violation of regulatory requirements was identified because subsequent testing by the licensee determined that the valves were functional.

  The finding would become a more significant safety concern, if left uncorrected, and was, therefore, more than a minor concern. Specifically, the failure to correctly evaluate a degraded/nonconforming condition potentially affecting the operability of structures, systems, and components (SSCs) required to be operable by TS could reasonably result in an unrecognized condition of a SSC failing to fulfill a safety-related function. Because the SPMU system was primarily associated with long term decay heat removal following certain design basis accidents, the inspectors concluded that this issue was associated with the Mitigating Systems Cornerstone. The finding was of very low safety significance because the issue: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event. The inspectors concluded that this finding affected the cross-cutting area of human performance because the licensee did not have a formal process in place with adequate guidance and training to enable licensed senior reactor operators to properly and promptly evaluate operability in this instance. As a result, senior reactor operators took it for granted that utilizing the relief allowed by TS Surveillance Requirement 3.0.3 and performing a risk evaluation obviated the need to address operability of the instrumentation supported by the excess flow check valves for the ASME/ANSI Code noncompliance. (IMC 0310 H.1(a)) (Section 1R15.b.(1))

Cornerstone: Barrier Integrity

- Severity Level IV. The inspectors identified a finding of very low safety significance with an associated NCV of 10 CFR 50.59, "Changes, Tests and Experiments." The licensee failed to perform an adequate 10 CFR 50.59 evaluation and obtain a license amendment prior to implementing CPS 3711.01, "CPS [Clinton Power Station] Operations with the Potential to Drain the Reactor Vessel [OPDRV]," Revision 0 on January 11, 2010. The procedure established a definition of an OPDRV for use in determining the applicability of several TS requirements while in Modes 4 and 5. The licensee failed to recognize that implementing this new procedure, in effect, constituted a change to the TS incorporated into its licensing basis, which would, therefore, require a license amendment pursuant to 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90. No immediate corrective actions were taken to address this finding; however, the licensee entered this issue into its corrective action program for evaluation.

  The finding was of more than minor significance because there was a reasonable likelihood that the change requiring a 10 CFR 50.59 evaluation would require NRC review and approval prior to implementation. Because this issue affected the

NRC's ability to perform its regulatory function, the inspectors evaluated it using the traditional enforcement process and assessed the significance of the underlying issue using the SDP. Based on the results of a modified Phase 2 SDP evaluation, this finding was determined to be of very low safety significance. Consistent with the guidance in Supplement I, Paragraph D.5, of the NRC Enforcement Policy, the violation associated with this finding was determined to be a Severity Level IV Violation. The inspectors concluded that this finding affected the cross-cutting area of human performance. Specifically, the licensee did not use conservative decision making to demonstrate that the proposed action did not require prior NRC approval. The inspectors noted that the licensee was aware of potential concerns regarding the new procedure prior to completing the initial 10 CFR 50.59 evaluation and again prior to revising the evaluation in response to concerns raised by the inspectors; however, the incorrect conclusion was reached in both revisions of the evaluation that the new procedure was not a change to the TS and that a license amendment was not necessary. (IMC 0310 H.1(b)) (Section 1R13.b.(1))

- Green. A finding of very low safety significance with an associated NCV of 10 CFR 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings,” was self-revealed on January 29, 2010, when the dryer cavity gate seal depressurized during the performance of the containment and reactor vessel isolation functional surveillance procedure. When the seal lost pressure, approximately 46,500 gallons of water leaked from the dryer cavity pool into the reactor cavity. In response to the event, the licensee ensured all personnel were out of the reactor cavity, entered its radioactive spill off-normal procedure, and re-established air pressure to the dryer cavity gate seal. Subsequent investigation revealed that during the gate seal inflation procedure, the proper valve operation sequence was not followed. As a corrective action, the licensee revised many of its procedures and included a special brief to the refueling outage preparation for Reactor Services personnel.

  The finding was of more than minor significance because the licensee’s failure to correctly install the upper containment dryer cavity gate could be reasonably viewed as a precursor to a significant event and, if left uncorrected, would potentially lead to a more significant safety concern (i.e., increased dose or personnel contamination). In addition, the finding was similar to Example 4c in IMC 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues,” in that, data recorded during installation of the dryer cavity gate seal was incorrect and resulted in backup air bottle supply pressure left outside the acceptable range. Because the dryer cavity gate seal is intended to contain highly radioactive fluids within containment, which supports the radiological barrier functions to protect plant workers and the public following serious transients or accidents, the inspectors concluded that this issue was associated with the Barrier Integrity Cornerstone. Although this event resulted in a loss of inventory from the dryer cavity pool and partial flooding of the lower reactor cavity and drywell, it was determined to be of very low safety significance because there was no loss of inventory from the reactor vessel and it could not result in the loss of reactor coolant system level instrumentation. The inspectors concluded that this finding affected the cross-cutting area of human performance. The licensee's Root Cause Report described the root cause as the maintenance craftsman performed steps out of sequence and failed to comply with the procedure. Therefore, as concluded by the Root Cause, in this instance, the licensee did not effectively communicate expectations regarding procedural compliance and, as a result, the Reactor Services maintenance craftsman did not correctly follow the procedure by performing steps out of sequence and restoring a system to service that was incorrectly aligned. (IMC 0310 H.4(b)) (Section 4OA2.3.b.(1))

Cornerstone: Emergency Preparedness

- Green. The inspectors identified a finding of very low safety significance with an associated NCV of 10 CFR 50.54(q) for the licensee's failure to follow and maintain the Emergency Plan, which meets the standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR 50. Specifically, the licensee's Emergency Plan calls for the performance of periodic drills to evaluate the ability to augment its Emergency Response Organization (ERO). However, the Emergency Plan implementing procedure used for the conduct of these augmentation drills exempts certain ERO members from participation in these drills, a situation which prevents the licensee from fully demonstrating its ability to augment all the ERO positions in a timely manner. The licensee's approved Emergency Plan does not provide for such an exemption. The licensee entered the finding into the corrective action program.

  The use of an implementing procedure that causes the conduct of an activity to be inconsistent with the associated requirements in the licensee's Emergency Plan results in a failure to follow and maintain the Emergency Plan and is a performance deficiency. As a result of the limitations in the procedure, the licensee failed to conduct call-in drills to demonstrate timely augmentation of ERO positions filled by skilled/technical personnel. The deficiency did not impact the NRC's regulatory process or contribute to actual safety consequences; therefore, the performance deficiency was screened using the Emergency Preparedness Significance Determination Process as a failure to comply. The deficiency was determined to be more than minor because the deficiency adversely affected the Emergency Preparedness Cornerstone objective and had the attribute associated with ERO readiness in the area of ERO augmentation testing. The inspector evaluated the finding using the IMC 0609, Appendix B, Sheet I, "Failure to Comply" Flowchart. The inspector evaluated the finding as a degraded planning standard function since the licensee's conduct of the augmentation exercises did not include all ERO positions. The finding was determined to be of very low safety significance. The inspector determined the finding had a cross-cutting aspect in the problem identification and resolution area with a component in self and independent assessments. The licensee's augmentation call-in drills were not comprehensive to include all ERO augmentation staffing positions. (IMC 0310 P.3(a)) (Section 1EP3.1)

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

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

REPORT DETAILS

Summary of Plant Status

Unit 1 was operated at or near full power during the inspection period with one exception. On May 16, 2010, the licensee reduced power to about 71 percent to perform control rod pattern adjustments and main steam isolation valve channel functional testing. The licensee returned the unit to full power later the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R01 Adverse Weather Protection (71111.01)

.1 Readiness For Impending Hot Summer Weather Conditions

a. Inspection Scope

The inspectors evaluated the licensee's preparations for hot summer weather conditions, focusing on the auxiliary power system and the control room ventilation system. During the week of April 19, 2010, the inspectors performed a detailed review of severe weather and plant de-winterization procedures and performed general area plant walkdowns. The inspectors focused on plant-specific design features and implementation of procedures for responding to or mitigating the effects of hot summer weather conditions on the operation of the plant. The inspectors reviewed system health reports and system engineering summer readiness review documents for the above systems.

Additionally, the inspectors reviewed selected action requests for the identification and resolution of procedure and equipment deficiencies associated with adverse weather mitigation.

This inspection constituted one seasonal extreme weather readiness inspection sample as defined in Inspection Procedure (IP) 71111.01.

b. Findings

No findings of significance were identified.

.2 Summer Readiness of Offsite and Alternate Alternating Current (AC) Power Systems

a. Inspection Scope

The inspectors evaluated the licensee's plant features and procedures for operation and continued availability of offsite and alternate AC power systems. The inspectors interviewed plant personnel and reviewed the licensee's communications protocols between the Transmission System Operator (TSO) and the plant to verify that the appropriate information was being exchanged when issues arose that could impact the offsite power system. Aspects considered in the inspectors' review included:

- The actions to be taken when notified by the TSO that the post-trip voltage of the offsite power system at the plant will not be acceptable to assure the continued operation of the safety-related loads without transferring to the onsite power supply;

- The compensatory actions identified to be performed if it is not possible to predict the post-trip voltage at the plant for the current grid conditions;
- The required re-assessment of plant risk based on maintenance activities that could affect grid reliability, or the ability of the transmission system to provide offsite power; and
- The required communications between the plant and the TSO when changes at the plant could impact the transmission system, or when the capability of the transmission system to provide adequate offsite power is challenged.

The inspectors performed a walkdown of the switchyard with a plant operator and a TSO electrician to observe the material condition of the offsite power sources. The inspectors also reviewed the status of outstanding work orders to assess whether corrective actions for any degraded conditions were scheduled with the TSO with the appropriate priority.

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

b. Findings

No findings of significance were identified.

.3 Readiness For Impending Adverse Weather Condition – Tornado/High Winds

a. Inspection Scope

Since thunderstorms with potential tornados and high winds were forecast in the vicinity of the facility for the week of April 5, 2010, the inspectors reviewed the licensee's overall preparations/protection for the expected conditions. The inspectors toured the plant grounds in the vicinity of the main power transformers, unit auxiliary transformers, reserve auxiliary transformers, emergency reserve auxiliary transformer, and static VAR compensators to look for loose debris, which, if present, could become missiles during a tornado or with high winds. During the inspections, the inspectors focused on plant-specific design features and the licensee's procedure used to respond to tornado and high winds conditions.

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

b. Findings

No findings of significance were identified.

1R04 Equipment Alignment (71111.04)

.4 Quarterly Partial System Walkdowns (71111.04Q)

a. Inspection Scope

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

- Reactor Core Isolation Cooling (RCIC) System (single train risk-significant system;
- Division 2 Emergency Diesel Generator (DG) during maintenance on Division 1 safety systems;
- Residual Heat Removal (RHR) Train 'B' during maintenance on RHR Train 'A.'

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 with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted three partial system walkdown inspection samples as defined in IP 71111.04.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05)

.1 Routine Resident Inspector Tours (71111.05Q)

a. Inspection Scope

The inspectors performed fire protection tours in the following plant areas:

- Fire Zone A3-g, “Containment Electrical Penetration Area (West) – Elevation 781'-0";
- Fire Zones CB-3b & 3d, “Division 4 NSPS [Nuclear System Protection System] Inverter and Battery Rooms – Elevation 781'-0";
- Fire Zone F-1i, “Fuel Pool Cooling Pump Room - Elevation 712'-0";
- Fire Zone F-1n, “Fuel Pool Cooling Heat Exchanger Room - Elevation 737'-0"; and
- Fire Zone R-1n, “Paint and Oil Storage Room - Elevation 737'-0."

The inspectors verified that transient combustibles and ignition sources were appropriately controlled and assessed the material condition of fire suppression systems, manual firefighting equipment, smoke detection systems, fire barriers and emergency lighting units. 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; that the licensee’s fire plan was in alignment with actual conditions; and that fire doors, dampers, and penetration seals appeared to be in satisfactory condition.

In addition, the inspectors verified that fire protection related problems were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted five quarterly fire protection inspection samples as defined in IP 71111.05AQ.

b. Findings

No findings of significance were identified.

1R06 Flooding Protection Measures (71111.06)

.1 Unanalyzed Condition of Interconnecting Floor Drains Between the RHR 'A' Pump Room and Radwaste Pipe Tunnel

(Closed) Unresolved Item (URI) 05000461/2009004-01, "Interconnecting Floor Drains Between the RHR 'A' Pump Room and Radwaste Pipe Tunnel"

a. Inspection Scope

The inspectors had previously identified that floor drains in the RHR 'A' Pump Room and the Radwaste Pipe Tunnel were interconnected, which resulted in an unanalyzed condition. This issue was reviewed by the inspectors during this inspection period to resolve open questions regarding the licensee's evaluation of the extent of condition and the reporting requirements. The extent of condition involved interconnecting floor drains between the RHR 'C' Pump Room and the Auxiliary Building Floor Drain Tank Room and Pump Room.

The inspectors discussed the licensee's evaluation of the extent of condition and the reporting requirements with the licensee. The inspectors reviewed the flood analyses and design documents, including the Updated Final Safety Analysis Report (UFSAR), plant drawings, and engineering calculation IP-M-0782, "Suppression Pool Equalization Levels," Revision 0.

This inspection constituted one internal flood protection inspection sample as defined in IP 71111.06.

b. Findings

(1) Failure to Satisfy 10 CFR 50.72 and 50.73 Reporting Requirements

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated Severity Level IV non-cited violation of the NRC's reporting requirements in 10 CFR 50.72(a)(1), "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 10 CFR 50.73(a)(1), "Licensee Event Report System." The licensee failed to make a required 8-hour non-emergency notification call to the NRC Operations Center and also failed to submit a required Licensee Event Report (LER) within 60 days after discovery of a condition that resulted in the plant being in an unanalyzed condition that significantly degraded plant safety and could have prevented fulfillment of the safety function of the emergency core cooling system (ECCS).

Discussion

During review of plant drawings for floor drain system piping in the ECCS and RCIC Pump Rooms on the 707'0" elevation of the Auxiliary Building, the inspectors identified that floor drains in the RHR 'A' Pump Room appeared to be connected via permanent 4-inch pipe embedded in the floor to floor drains in the Radwaste Pipe Tunnel, which is located along the western wall of the (adjacent) Control Building at the 720'0" elevation. The inspectors noted that each of the separate pump rooms was supposedly designed to be isolated from other areas of the plant and not susceptible to flooding from sources external to the pump rooms.

The inspectors discussed this floor drain configuration with the licensee and questioned the adequacy of the design with respect to the potential for flooding. The inspectors noted that Section 3.8.4.1.1 of the UFSAR stated that the ECCS Pump Rooms are in flood protection compartments with watertight doors. In the event of a pipe rupture, the flooding in one compartment will not result in the flooding of any other compartment, and the failure of a pump suction line will not drain the suppression pool. Section D3.6.4 of the UFSAR stated that a postulated failure of any of the non-isolable portions of the ECCS pump suction lines to the suppression pool could result in flooding of a single ECCS cubicle to the high water level in the suppression pool (731'5" elevation). Due to the interconnecting floor drain piping, if flooding in the RHR 'A' Pump Room (from the suppression pool) were to occur, then the potential existed that cross-flooding could occur between the RHR 'A' Pump Room and the Radwaste Pipe Tunnel. Flooding could potentially continue until the suppression pool level was below the Control Building floor drain level (720'6" elevation).

The inspectors discussed this issue in greater detail and documented a finding with an associated NCV of 10 CFR 50, Appendix B, Criteria III, "Design Control," in NRC Inspection Report 05000461/2010002, regarding the licensee's failure to correctly translate the design basis into the design of the Auxiliary Building floor drain system with appropriate margin. To address the immediate operability concern, the licensee plugged the two floor drains in the Radwaste Pipe Tunnel to prevent communication with the floor drain system in the RHR 'A' Pump Room. The exposed vertical section of the drain line in the low pressure core spray (LPCS) Pump Room was then cut and a solid steel plate welded into the pipe per an engineering design change to permanently isolate the floor drains between the two rooms.

The inspectors reviewed the licensee's evaluation of the as-found condition with the interconnecting floor drains between the RHR 'A' Pump Room and Radwaste Pipe Tunnel. The licensee concluded in its evaluation that, while the interconnecting floor drains represented an unanalyzed condition, that was not consistent with the licensing basis, safe shutdown would still be assured. The licensee evaluated the concern with potential flooding in the RHR 'A' Pump Room in terms of 10 CFR 50, Appendix A, "General Design Criteria." For a pipe break, General Design Criteria (GDC) 4, "Environmental and Missile Design Bases," and NUREG-800, "Standard Review Plan," require that the break be evaluated coincident with the functional failure of any single active component, a seismic event the level of the safe shutdown earthquake, and a loss of offsite power. This is consistent with UFSAR Section 3.6.1.3.1, which states, in part, that “[i]n the plant design, consideration was given to the effects of postulated piping breaks with respect to the limits of acceptable damage/loss of function, to assure that, even with coincident single loss of active component, and earthquake equal to the safe shutdown earthquake, and loss of offsite power, the remaining structures, systems, and components would be adequate to safely shutdown the plant.” The RHR ‘A’ pump has multiple safety functions, and, consistent with the design basis assumptions, all of these functions would be lost in the event of a pipe break in the pump room. The suppression pool supports the functions of all three divisions of the ECCS and, if the suppression pool were impacted by a single failure, that single failure could render multiple divisions incapable of performing their design safety functions. That would be contrary to the current licensing basis. The suppression pool also supports the pressure suppression function of the primary containment. If the suppression pool level were to drain below the minimum vent cover level of 15’1” (727’1” elevation), an insufficient amount of water would be available to adequately condense the steam from the safety/relief valve quenchers, main vents, or RCIC turbine exhaust lines.

During review of the licensee’s evaluation and discussion with engineering staff, the inspectors identified that a postulated failure of the 20” suction piping between the remote manual containment isolation valve, 1E12-F004A, and the RHR ‘A’ pump under the requirements of GDC 4 would result in non-isolable flooding into the RHR ‘A’ Pump Room, and hence into the Radwaste Pipe Tunnel through the floor drains, that would drain the suppression pool below the suppression pool high water level assumed in Section D3.6.4 of the UFSAR since the flood level in the room would not equalize with the suppression pool above the suppression pool high water level. This particular scenario had not been considered by the licensee in the evaluation. This portion of the piping system is within the moderate-energy portion of the fluid system. The licensee determined that a break or crack in the line, when calculated per UFSAR Section 3.6.2.1.5.b, would result in a 206 gallons-per-minute (gpm) leak.

In its evaluation of the unanalyzed condition, the licensee concluded that it was not reportable to the NRC under the requirements of 10 CFR 50.72(b)(3)(ii)(B) or 50.73(a)(2)(ii)(B) as a condition that results (or resulted) in the nuclear power plant being in an unanalyzed condition that significantly degrades (or degraded) plant safety. The licensee also concluded that the non-conforming condition was not reportable to the NRC under the requirements of 10 CFR 50.72(b)(3)(v)(D) or 50.73(a)(2)(v)(D) as a condition that could have prevented the fulfillment of the safety function of structures or systems needed to mitigate the consequences of an accident. The inspectors questioned this conclusion. The licensee wrote action request (AR) 01031977 to address the open questions and subsequently submitted a voluntary LER (LER 05000461/2010-001-00, “Unanalyzed Leakage Pathway Affecting Residual Heat Removal ‘A’ Pump Room Flooding Analysis”). However, the LER was submitted after the 60-day reporting period and, as discussed below, the licensee reached an incorrect conclusion in the LER because it did not take into consideration those safety functions that would still be required after safe shutdown has initially been achieved.

During this inspection period, the inspectors concluded that the unanalyzed condition met the 10 CFR 50.72(b)(3)(ii)(B), 50.73(a)(2)(ii)(B), 50.72(b)(3)(v)(D), and 50.73(a)(2)(v)(D) reporting criteria. According to the licensee’s evaluation, the postulated sequence of events for the above non-isolable pipe break scenario would progress slowly and safe shutdown could be achieved after about 11 hours. However, the inspectors concluded that the postulated event would still significantly degrade plant safety and it could still result in the loss of safety function of the low pressure injection/spray ECCS subsystems. Technical Specification 3.5.2 requires that two ECCS injection/spray subsystems be operable in Modes 4 and 5, except with the reactor cavity to steam dryer pool gate removed and water level ≥ 22'8" over the top of the reactor pressure vessel flange. Assuming the suppression pool continues to drain into the Radwaste Pipe Tunnel, with suppression pool level below the TS 3.5.2 minimum level of 12'9" (724'9" elevation), the low pressure injection/spray ECCS subsystems would not be capable of fulfilling their safety function. According to Emergency Operating Procedure (EOP)-6, Table Z, "NPSH [Net Positive Suction Head] / Vortex Limits," the minimum suppression pool level required to maintain NPSH to the ECCS pumps is 11'0". While operators may be able to bring the plant to safe shutdown before the suppression pool reaches the minimum vent coverage level of 15'1", the suppression pool would still continue to drain due to the non-isolable pipe break and, without the addition of water, would eventually reach the level that would not support the ECCS pumps. The inspectors concluded that by this point in the scenario, plant safety would be significantly degraded. There are available makeup sources of water for the suppression pool, even with a loss of offsite power. Dumping the upper containment pool to the suppression pool and/or addition of water from the RCIC storage tank using the high pressure core spray pump would provide additional time before reaching 12'9" (or 11'0") for the licensee to locate the two floor drains in the Radwaste Pipe Tunnel and plug them to stop the suppression pool drain down. However, plugging floor drains in this infrequently accessed plant area would involve operator manual actions not considered or credited in the current licensing basis.

Because the condition existed (i.e., it was occurring) at the time of discovery on October 7, 2009, the inspectors concluded that the 10 CFR 50.72(b)(3) criteria for an 8-hour notification call to the NRC were met and the licensee should have made the appropriate notification call. The licensee corrected the floor drain condition on the following day, October 8, 2009.

Analysis

The inspectors determined that the licensee's failure to report this issue as a condition that results (or resulted) in the nuclear power plant being in an unanalyzed condition that significantly degrades (or degraded) plant safety, and as a condition that could have prevented the fulfillment of the safety function of structures or systems needed to mitigate the consequences of an accident, was a licensee performance deficiency warranting a significance evaluation. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports, Appendix B, "Issue Screening," because these violations of the NRC's reporting requirements affected the NRC's ability to perform its regulatory function, the inspectors evaluated the violations using the traditional enforcement process in accordance with the NRC Enforcement Policy and assessed the significance of the underlying issue using the SDP. The inspectors reviewed the examples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and found no examples related to this issue. The inspectors determined that this finding was of more than minor significance because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the TS and the regulations in order to perform its regulatory function. The inspectors previously determined that the underlying issue was a finding of very low safety significance (Green) during a Phase 3 SDP review and documented the finding in NRC Inspection Report 05000461/2010002 (NCV 05000461/2010002-03, Unanalyzed

Condition of Interconnecting Floor Drains Between the RHR ‘A’ Pump Room and Radwaste Pipe Tunnel). Consistent with the guidance in Supplement I, Paragraph D.4, of the NRC Enforcement Policy, the violation associated with this finding was determined to be a Severity Level IV Violation.

Cross-Cutting Aspects

The inspectors concluded that this finding affected the cross-cutting area of human performance. Specifically, the licensee did not use conservative assumptions in decision making while evaluating the reportability of the unanalyzed condition with respect to the reporting requirements in 10 CFR 50.72(a)(1)(ii) and 50.73(a)(1). Specifically, the licensee did not take into consideration those safety functions that would still be required after safe shutdown has initially been achieved. (IMC 0310 H.1(b))

Enforcement

Title 10 CFR 50.72(a)(1)(ii) required, in part, that the licensee shall notify the NRC Operations Center via the Emergency Notification System of those non-emergency events specified in Paragraph (b) that occurred within three years of the date of discovery. Title 10 CFR 50.72(b)(3) required, in part, that the licensee shall notify the NRC as soon as practical and in all cases within eight hours of the occurrence of any of the applicable conditions. Title 10 CFR 50.72(b)(3)(ii)(B) required, in part, that the licensee report any event or condition that results in the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety. Title 10 CFR 50.72(b)(3)(v)(D) required, in part, that the licensee report any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident.

Title 10 CFR 50.73(a)(1) required, in part, that the licensee submit an LER for any event of the type described in this paragraph within 60 days after the discovery of the event. Title 10 CFR 50.73(a)(2)(ii)(B) required, in part, that the licensee report any event or condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety. Title 10 CFR 50.73(a)(2)(v)(D) required, in part, that the licensee report any event or condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident.

Contrary to the above:

1. The licensee failed to notify the NRC Operations Center via the Emergency Notification System of a non-emergency event specified in Paragraph (b) within eight hours after discovery of an event on October 7, 2009. The event involved the discovery of interconnecting floor drains between the RHR ‘A’ Pump Room and the Radwaste Pipe Tunnel, which resulted in the plant being in an unanalyzed condition that significantly degraded plant safety and at the time of discovery could have prevented fulfillment of the safety function of the ECCS.

2. The licensee failed to submit a required LER within 60 days after discovery of an event on October 7, 2009. The event involved the discovery of interconnecting floor drains between the RHR ‘A’ Pump Room and the Radwaste Pipe Tunnel, which

resulted in the plant being in an unanalyzed condition that significantly degraded plant safety and could have prevented fulfillment of the safety function of the ECCS.

This Severity Level IV violation of the NRC reporting requirements is associated with a Green SDP finding and will be treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000461/2010003-01, Failure to Satisfy 10 CFR 50.72 and 50.73 Reporting Requirements).** The licensee entered this finding into its corrective action program as AR 01080117.

(2) <u>Extent of Condition Review – Interconnecting Floor Drains Between the RHR 'C' Pump Room and the Auxiliary Building Floor Drain Tank Room and Pump Room</u>

During its extent of condition review, the licensee also discovered that a similar arrangement existed with floor drains on the west side of the Auxiliary Building in that the RHR 'C' Pump Room floor drain piping communicates with the floor drains in the Auxiliary Building Floor Drain Tank Room and Pump Room. Those rooms are located south of the RHR 'C' Pump Room on the other side of the watertight door, at the 712'0" elevation. The licensee evaluated this configuration and concluded that draining the suppression pool through this drain path in the event of a pipe break in RHR 'C' Pump Room was not a concern. The licensee noted that the combined volume of the RHR 'C' Pump Room, Auxiliary Building Floor Drain Tank Room, and Auxiliary Building Floor Drain Pump Room was about the same as the LPCS Pump Room. Therefore, the licensee concluded that in the event of a pipe break in RHR 'C' Pump Room, the suppression pool could not be drained below the equalization elevation for the LPCS Pump Room, which was already believed to be acceptable.

However, during review of the licensee's evaluation, the inspectors noted that the equalization level for the LPCS Pump Room, as stated in Table 2 of CPS 4304.01, "Flooding," Revision 4e, was 12'1". This level is well below the minimum vent cover level of 15'1" and also below the TS 3.5.2 minimum suppression pool level of 12'9" for operability of the ECCS subsystems with the unit in Modes 4 and 5. The inspectors questioned whether the results of the existing Flood Analysis for the LPCS Pump Room was acceptable based on having the equalization level below the minimum vent cover level in the event of a non-isolable suction pipe break and similarly whether the above evaluation for the RHR 'C' Pump Room was acceptable. In response to the inspectors' questions, the licensee wrote AR 01039042 to address apparent discrepancies with the suppression pool equalization levels in Table 2 of CPS 4304.01 and the possible impact on the Flood Analysis for the ECCS Pump Rooms. From this action request, the licensee identified the need to complete a formal calculation confirming the flooding equalization levels and issue it with an engineering evaluation that includes an operating procedure review and identification of any necessary operations training. This is important because licensed operators would utilize information from this table in evaluating actions to be taken under the emergency operating procedures in the event of ECCS pump room flooding.

During this inspection period, the inspectors reviewed the licensee's calculation and associated engineering evaluation and found no additional issues of concern. The equalization levels for the LPCS and RHR 'C' Pump Rooms are below the 15'1" minimum vent cover level at 14'4" and 14'10", respectively. However, the licensing basis does not assume a loss of coolant accident coincident with a non-isolable pump room flooding event and there is sufficient margin to the TS 3.5.2 minimum suppression pool level of 12’9” to support operability of the ECCS pumps. The equalization level for the RHR ‘C’ Pump Room was calculated to include the volume of the Auxiliary Building Floor Drain Tank Room and Pump Room because the interconnecting floor drains have been accepted as-is.

The inspectors concluded that the licensee’s failure to have an accurate supporting calculation for the flooding equalization levels in Table 2 of CPS 4304.01 constituted a violation of 10 CFR 50, Appendix B, Criteria III, “Design Control,” of minor significance and is not subject to enforcement action in accordance with the NRC’s Enforcement Policy. The licensee entered this violation into its corrective action program as AR 01039042.

Unresolved Item 05000461/2009004-01 is closed.

1R07 Heat Sink Performance (71111.07)

.1 Annual Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors reviewed the licensee’s maintenance activities for the RHR system. Specifically, the review included the program for testing and analysis of the RHR Pump Room Ventilation Cooling Unit (heat exchanger) and RHR Heat Exchanger Room Ventilation Cooling Unit, which were cleaned, inspected, and evaluated. The inspectors assessed the as-found and as-left condition of the heat exchangers by direct observation and document reviews to verify that no deficiencies existed that would adversely impact the heat exchangers' ability to transfer heat to the shutdown service water (SX) system and to ensure that the licensee was adequately addressing problems that could affect the performance of the heat exchangers. The inspectors observed portions of inspection and cleaning activities, and reviewed documentation to verify that the inspection acceptance criteria specified in procedure ER-AA-340-1002, “Service Water Heat Exchanger and Component Inspection Guide,” Revision 4, were satisfactorily met.

This inspection constituted two annual heat sink inspection samples as defined in IP 71111.07.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification Program (71111.11)

.1 Resident Inspector Quarterly Review (71111.11Q)

a. Inspection Scope

The inspectors observed licensed operators during simulator training on June 9, 2010. The inspectors assessed the operators’ response to the simulated events focusing on alarm response, command and control of crew activities, communication practices, procedural adherence, and implementation of Emergency Plan requirements.
The inspectors also observed the post-training critique to assess the ability of licensee evaluators and operating crews to self-identify performance deficiencies. 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 inspection sample as defined in IP 71111.11.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Quarterly Evaluations (71111.12Q)

a. Inspection Scope

The inspectors evaluated the licensee's handling of selected degraded performance issues involving the following risk-significant structures, systems, and components (SSCs):

- Feedwater System Primary Containment Isolation Valves 1B21F032A and 1B21F032B;
- Reactor Recirculation System Flow Control Valves 1B33F060A and 1B33F060B.

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the SSCs. Specifically, the inspectors independently verified the licensee's handling of SSC performance or condition problems in terms of:

- Appropriate work practices;
- Identifying and addressing common cause failures;
- Scoping of SSCs in accordance with 10 CFR 50.65(b);
- Characterizing SSC reliability issues;
- Tracking SSC unavailability;
- Trending key parameters (condition monitoring);
- 10 CFR 50.65(a)(1) or (a)(2) classification and reclassification; and
- Appropriateness of performance criteria for SSC functions classified (a)(2) and/or appropriateness and adequacy of goals and corrective actions for SSC functions classified (a)(1).

In addition, the inspectors verified that problems associated with the effectiveness of plant maintenance were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted two maintenance effectiveness inspection samples as defined in IP 71111.12.

b. Findings

No findings of significance 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 maintenance and emergent work activities affecting the risk significant and safety-related equipment listed below to verify that the appropriate risk assessments were performed prior to removing equipment for work:

- Planned maintenance on March 25th to drain the RHR ‘A’ Heat Exchanger;
- Planned maintenance and testing on April 22nd on the High Pressure Core Spray System;
- Planned maintenance on the Emergency Reserve Auxiliary Transformer (ERAT) for a system outage window the week of May 3rd.

These activities were selected based on their potential risk significance relative to the Reactor Safety Cornerstones. As applicable for each of the above activities, the inspectors reviewed the scope of maintenance work in the plant’s daily schedule, reviewed Control Room logs, verified that plant risk assessments were completed as required by 10 CFR 50.65(a)(4) prior to commencing maintenance activities, discussed the results of the assessment with the licensee’s Probabilistic Risk Analyst and/or Shift Technical Advisor, and verified that plant conditions were consistent with the risk assessment assumptions. The inspectors also reviewed TS requirements and walked down portions of redundant safety systems, when applicable, to verify that risk analysis assumptions were valid, that redundant safety related plant equipment necessary to minimize risk was available for use, and that applicable requirements were met.

The inspectors also completed a review of the licensee’s procedure for determining what constitutes an operation with the potential to drain the reactor vessel (OPDRV) and resolved open questions with the licensee’s 10 CFR 50.59 evaluation.

In addition, the inspectors verified that maintenance risk related problems were entered into the licensee’s corrective action program with the appropriate significance characterization. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted three maintenance risk assessment inspection samples as defined in IP 71111.13.

b. Findings

(1) Failure to Perform an Adequate 10 CFR 50.59 Evaluation For Clinton Power Station (CPS) Procedure 3711.01

(Closed) URI 05000461/2010002-06, "Questions Regarding 10 CFR 50.59 Evaluation for CPS Procedure 3711.01 Involving Operations with the Potential to Drain the Reactor Vessel (OPDRV)"

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated Severity Level IV non-cited violation of 10 CFR 50.59, "Changes, Tests and Experiments," for the licensee's failure to perform an adequate 10 CFR 50.59 evaluation and obtain a license amendment prior to implementing CPS 3711.01, "CPS Operations with the Potential to Drain the Reactor Vessel," Revision 0. Specifically, the licensee failed to recognize that implementing this new procedure, in effect, constituted a change to the TS incorporated into its licensing basis, which would, therefore, require a license amendment pursuant to 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.

Discussion

On January 11, 2010, the licensee approved procedure CPS 3711.01, which established how it chose to define an OPDRV. OPDRV is a term used in a number of TS, including CPS TSs 3.3.6.1, 3.6.1.2, 3.6.1.3, 3.6.4.1, 3.6.4.2, 3.6.4.3, 3.7.3, and 3.7.4. The term is also important in the action statements of TS Limiting Conditions for Operation (LCOs) 3.5.2, 3.8.2, and 3.8.10.

Notably, the new procedure changed how control rod drive mechanism (CRDM) replacement was classified by the licensee. Prior to the change, this evolution had been treated as an OPDRV; however, once the procedure was approved, the evolution was no longer considered an OPDRV. During the refueling outage that began the following day on January 12th, there was no longer a licensee procedural requirement to enter a number of TS LCO action statements during CRDM replacements.

In CPS 3711.01, the licensee, among other things, chose to define a maximum hole diameter size of 1.92 inches below normal water level, which was to be considered to have no potential to drain the reactor vessel. This number was based upon the licensee's ability to provide non-safetyrelated makeup measures to compensate for reactor coolant system (RCS) inventory loss and, therefore, maintain reactor vessel water level above the vessel flange. The licensee performed an evaluation (CL-2010-E-001, "CPS Procedure 3711.01," Revision 0) to address the requirements of 10 CFR 50.59 prior to implementing the new procedure. The licensee's evaluation concluded that the procedure change did not involve a revision to the TS and that the impact of the change could not be more than minimal; therefore, a license amendment was not required prior to its approval.

In the initial revision of CL-2010-E-001, the licensee stated that a 17-hour period is established to take action to restore reactor vessel water level before irradiated fuel could become uncovered based upon the 1.92 inch hole diameter size. After the inspectors identified potential event scenarios introduced by the procedure change that were not analyzed, the licensee revised the evaluation. In the revised evaluation, the licensee concluded that 6 hours and 58 minutes would be available to take action in a loss of offsite power scenario to restore water level before uncovering irradiated fuel. Despite the significantly less response time available (17 hours versus approximately 7 hours), the licensee again concluded that a license amendment request was not required. In response to additional questions from the inspectors, the licensee also analyzed a possible scenario involving the removal of a control rod that is relied upon to isolate leakage for a removed CRDM. In this instance, the licensee concluded that 7 hours and 36 minutes would be available to respond to the event.

No detailed discussion was provided in either version of the 10 CFR 50.59 evaluation as to how station personnel would respond to a potential drain down event in the absence of any procedural guidance. In the initial evaluation, the licensee generally assumed that the 17 hours considered to be available for event response would be sufficient. Notably, this initial evaluation did not attempt to credit the availability of the ECCS as a makeup source; however, the revised evaluation did. Technical Specification 3.5.2, "ECCS – Shutdown," contains the requirements for the operability of the ECCS subsystems while shutdown. In the applicability statement, it states that the TS does not apply when in Mode 5 with water level greater than 22' 8" over the reactor vessel flange and the gate removed between the reactor cavity and the steam dryer pool. Therefore, under the possible event conditions during refueling operations, there would be no TS requirement for any ECCS subsystem to be operable. The licensee, however, credited the ECCS in its revised evaluation for each possible event scenario by stating that an ECCS subsystem and an emergency diesel generator was required by Clinton procedure OU-CL-104, "Shutdown Safety Management Program Clinton Power Station." While this procedure did state in Step 4.3.3.5 that as a general guideline at least one ECCS pump should be available anytime the cavity gate is removed, it was not a requirement. In Attachment 2 of the procedure under the inventory control requirements for Mode 5 with the reactor cavity flooded, the procedure clearly stated that this was a requirement only for a "Green" shutdown risk condition. A "Green" shutdown risk condition is a condition of the least or minimal plant risk. For any other inventory control shutdown risk condition entered by the licensee, it would be incorrect to credit the availability of the ECCS for response to these possible event scenarios.

Technical Specification 3.5.2 contains the action statement to "suspend all operations that have a potential for draining the reactor vessel" under the applicable conditions. Since the conversion of CPS from TS to improved Standard Technical Specifications (NUREG-1434, "Standard Technical Specifications – General Electric Plants, BWR/6," Revision 1), the respective LCO action statement now directs the licensee to "suspend operations with a potential for draining the reactor vessel." During this conversion of TS, no discussion was documented in the CPS licensing basis regarding this difference in phrasing of the respective LCO action statements; therefore, the meaning of the action statements necessarily remains unchanged. Plain language understanding of "suspend all operations that have a potential to drain the reactor vessel" does not imply the inclusion of any qualifying statements and criteria such as those included in CPS 3711.01 prior to entering those corresponding TS LCO action statements. In addition to an un-isolable hole diameter size of 1.92 inches, many other criteria and conditions were assigned by the licensee before an evolution could be considered an OPDRV. This new process by which the licensee chose to define OPDRV is in contrast to the plain language that is the CPS licensing basis. Therefore, implementation of CPS 3711.01 involved a change to the TS that required prior approval of the NRC.

Analysis

The inspectors determined that the licensee's failure to perform an adequate 10 CFR 50.59 evaluation and obtain a license amendment prior to implementing CPS 3711.01 was a licensee performance deficiency warranting a significance evaluation. Consistent with the guidance in IMC 0612, "Power Reactor Inspection Reports, Appendix B, "Issue Screening," because this issue affected the NRC's ability to perform its regulatory function, the inspectors evaluated it using the traditional enforcement process and assessed the significance of the underlying issue using the SDP. The inspectors reviewed the examples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and found no examples related to this issue. The inspectors determined that this finding was of more than minor significance because there was a reasonable likelihood that the change requiring a 10 CFR 50.59 evaluation would require Commission review and approval prior to implementation.

The inspectors and the Regional Senior Reactor Analyst used IMC 0609, Appendix G, "Shutdown Operations Significance Determination Process," to evaluate the significance of this finding. The applicable Phase 1 checklist was Checklist 7, "BWR [Boiling Water Reactor] Refueling Operation with RCS level > 23'". The finding did not screen as very low safety significant in Phase 1 because it was a finding that increased the likelihood of a loss of RCS inventory. A modified Phase 2 SDP evaluation was conducted using Worksheet 3, "SDP Worksheet for a BWR Plant – Loss of Inventory in [Plant Operating State] POS-3 (Cavity Flooded)" and the associated event tree in Figure 4, "Event Tree for Loss of Inventory – BWR POS-3."

The Phase 2 worksheet was modified because it generally assumes that the RHR system is available for decay heat removal and that a loss of inventory would threaten the decay heat removal functions. During this period of CRDM maintenance, the common suction line for RHR shutdown cooling was isolated and the fuel pool cooling system was providing the decay heat removal function. In addition, the worksheet does not anticipate that auto initiation of ECCS would be available during cavity flooded conditions. Therefore, the analysis needs to account for this deviation from the methodology.

The first sequence that needs to be solved (labeled Sequence 3) involves loss of inventory (LOI), recovery of RHR in shutdown cooling (RHRREC) and long term makeup (LCOOL). Because the common suction line for RHR is in maintenance, the RHR system cannot be recovered. However, in this sequence level is recovered by successful short term injection. This level recovery allows decay heat removal via the fuel pool cooling system to be recoverable. Long term inventory source can be from multiple sources including RHR service water, condensate storage tank, demineralized water tank, the suppression pool and the condenser hotwell. Because the sequence involves successful injection into the RCS in the near term, a long time is available before the core cooling is interrupted by depletion of the source of water for the injection systems (the suppression pool is assumed to be the source of water for this near term injection).

The other sequences in the Phase 2 worksheet were solved assuming that approximately 7 hours was available to isolate the drain path before the level in the reactor cavity reached the reactor flange. The licensee provided information that both

RHR trains and at least one train of LPCS were available for automatic injection in the event of a loss of RCS inventory. These systems could also be started manually if the automatic initiation were to fail. Adequate level indication was available to identify a loss of RCS inventory event.

This finding was determined to be a "precursor finding," as defined in IMC 0609, Appendix G. Table 2 was used to estimate the Initiating Event Likelihood (IEL) for a LOI precursor. Given 7 hours to isolate, adequate level indication, and the ability to identify the leak path and stop it, the estimated IEL was "4." Maximum mitigation credit was applied for the RCS injection (both automatic and manual injection were available), and long term cooling functions because of the amount of time available for operators to respond and the number of RCS injection systems available. No credit was given for the isolation function given that this drain path would not be automatically isolated. This is conservative, given that the drain path is isolable with reasonable operator action, but is accounted for in the estimated IEL. Given these assumptions, three sequences of "8" were estimated, which is equivalent to a "7," or a change in core damage frequency of 1E-7/year. The dominant sequence is a loss of inventory event followed by the failure of RCS injection.

Based on the results of the modified Phase 2 SDP evaluation, this finding was determined to be of very low safety significance (Green). Consistent with the guidance in Supplement I, Paragraph D.5, of the NRC Enforcement Policy, the violation associated with this finding was determined to be a Severity Level IV Violation.

Cross-Cutting Aspects

The inspectors concluded that the performance characteristic of this finding, that is the most significant contributor to the performance deficiency, was in the cross-cutting area of human performance. Specifically, the licensee did not use conservative decision making before proceeding. The inspectors noted that the licensee was aware of potential concerns regarding the new procedure prior to completing the initial 10 CFR 50.59 evaluation and again prior to revising the evaluation in response to concerns raised by the inspectors; however, the incorrect conclusion, that the new procedure was not a change to the TS and that a license amendment was not necessary, was reached in both revisions of the evaluation. (IMC 0310 H.1(b))

Enforcement

Title 10 CFR 50.59, "Changes, Tests and Experiments," establishes conditions under which licensees may make changes to their facilities and procedures as described in the safety analysis report without prior NRC approval. Title10 CFR 50.59 (c)(1)(i) stated, in part, that a licensee may make changes in the facility as described in the final safety analysis report without obtaining a license amendment pursuant to 10 CFR 50.90 only if a change to the TS incorporated in the license is not required.

Contrary to the above, on June 15, 2009, the licensee implemented procedure CPS 3711.01, "CPS Operations with the Potential to Drain the Reactor Vessel," Revision 0, without obtaining prior NRC approval. Implementing this new procedure, in effect, constituted a change to the TS incorporated into the licensing basis by altering the plain language definition of an OPDRV in the TS and resulted in a significant change in process for when the licensee enters a number of TS LCO action statements.
This Severity Level IV Violation is associated with a Green SDP finding and will be

treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy. **(NCV 05000461/2010003-02, Failure to Perform an Adequate 10 CFR 50.59 Evaluation for CPS Procedure 3711.01)**. This violation was entered into the licensee's corrective action program as AR 01063405.

Unresolved Item 05000461/2010002-06 is closed.

1R15 Operability Evaluations (71111.15)

.1 Operability Evaluations

a. Inspection Scope

The inspectors reviewed the following issues:

- AR 01017464, "1B21F028A: LLRT [Local Leak Rate Test] on Main Steam Line A, B, and C Test Failure";
- AR 01065232, "0AP03E: ERAT: Core Ground Test Results Lower Than Expected";
- AR 01059174, "The ERDS [Emergency Response Data System] System Did Not Connect to the NRC";
- AR 01063878, "NRC TIA [Task Interface Agreement] Applicability of TSSR [Technical Specification Surveillance Requirement] 3.0.3 for IST [Inservice Testing] of Excess Flow Check Valves."

The inspectors selected these potential operability issues based on the risk significance of the associated components and systems. The inspectors verified that the conditions did not render the associated equipment inoperable or result in an unrecognized increase in plant risk. When applicable, the inspectors verified that the licensee appropriately applied TS limitations, appropriately returned the affected equipment to an operable status, and reviewed the licensee's evaluation of the issue with respect to the regulatory reporting requirements. 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 evaluation.

In addition, the inspectors verified that problems related to the operability of safety-related plant equipment were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

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

.2 Findings

(1) Operability Assessment of Inservice Testing Surveillance Discrepancies for Excess Flow Check Valves

(Closed) URI 05000461/2009003-04, "Review of Applicability of TSSR 3.0.3 to Multiple Missed Surveillance Intervals for Excess Flow Check Valves"

Introduction

The inspectors identified a finding of very low safety significance (Green) associated with the licensee's failure to evaluate the functionality of multiple excess flow check valves that had not been tested in accordance with the American Society of Mechanical Engineers / American National Standards Institute (ASME/ANSI) Code Inservice Testing (IST) requirements to establish whether the nonconforming condition warranted starting the TS action time for the suppression pool makeup (SPMU) system. No violation of regulatory requirements was identified because subsequent testing by the licensee determined that the valves were functional.

Discussion

On November 18, 2008, the licensee identified that nine excess flow check valves (1CM002A, 1CM002B, 1CM003A, 1E22-F330, 1E22-F332, 1E51-F377A, 1E51-F377B, 1SM008 and 1SM009) were incorrectly removed from its IST Program in 2002. The valves have a safety function to re-open following a design basis accident to provide instrumentation assumed to be available post-accident. The ASME/ANSI Operations and Maintenance Code (OMa 1988, Part 10) would require a position verification test for these valves once every two years and an opening test once every three months, with exceptions allowed for refueling cycle frequency. The valves had not been tested since 2000.

Upon discovery of the excess flow check valve testing issue, the licensee utilized the relief afforded by TSSR 3.0.3 for a missed surveillance to allow up to the limit of the specified frequency to perform missed surveillances. During review of the issue in April 2009, the inspectors questioned the licensee whether it was appropriate to utilize the relief allowed by TSSR 3.0.3 because these did not appear to be cases of a single missed surveillance.

The NRC staff had concluded in TIA 2008-004, "Evaluation of Application of TS 4.0.3, 'Surveillance Requirement Applicability,' at Pilgrim," that a missed surveillance (i.e., inadvertently exceeded surveillance) is not equivalent to a never-performed surveillance for which TSSR 3.0.3 would not apply. The basis for the relief allowed by TSSR 3.0.3 is that the past surveillance testing history provides a level of confidence that the component or system is most likely operable. A surveillance that has never been performed does not have this basis for a presumption of operability.

Consistent with the "level of confidence" argument that was provided in TIA 2008-004, the inspectors questioned whether it would be correct for the licensee to apply TSSR 3.0.3 for the excess flow check valves. After all, the licensee removed the valves from its IST Program and discontinued testing, exceeding four previously defined test frequency periods without testing the valves. Therefore, the basis for a presumption of operability may not exist because the licensee was not demonstrating operability by performing the required testing of the excess flow check valves all along. The inspectors open URI 05000461/2009003-04 to track the NRC staff's review of this question to determine if additional NRC guidance was necessary to specify whether TSSR 3.0.3 applies in the case where more than one surveillance interval is exceeded.

The inspectors used the "level of confidence" argument discussed in TIA 2008-004 as the basis to question the functionality of the valves. Up to this point, the licensee had taken for granted that utilizing the relief allowed by TSSR 3.0.3 and performing a risk evaluation obviated the need to address operability of the instrumentation supported by the excess flow check valves for the ASME/ANSI Code noncompliance. Subsequently, on June 20, 2009, the licensee revised the calculation defining the design basis function for the excess flow check valves to remove the active support safety function of five of the check valves. Of the remaining four check valves that have an active safety function (1CM002B, 1E22-F332, 1E51-F377B, and 1SM008), one check valve (1E22-F332) was tested satisfactorily with the unit in Mode 1 on March 6, 2009, and again on June 5, 2009. In response to the inspectors' questions, on June 24, 2009, the licensee completed an evaluation for the remaining three check valves. The inspectors subsequently reviewed the evaluation and discussed it with the licensee's staff. The inspectors' review of the operability evaluation was documented in NRC Inspection Report 05000461/2009-004. The inspectors did not believe that the licensee's supporting basis provided a high degree of confidence that the valves would function as required; however, without actually testing the valves, the inspectors were unable to prove that the valves would not function. The licensee maintained that it was satisfied with its determination in the evaluation. The licensee subsequently tested the three excess flow check valves satisfactorily during an unplanned forced outage that began on September 29, 2009. A licensee-identified NCV of 10 CFR 50.55a(f) was documented in NRC Inspection Report 05000461/2009-004 for the licensee's failure to perform inservice testing of the excess flow check valves.

The NRC staff reviewed the question of applicability of TSSR 3.0.3 to missed inservice testing of the excess flow check valves at Clinton Power Station in TIA 2010-001, "Evaluation of Application of Technical Specification Surveillance Requirement 3.0.3, 'Surveillance Requirement Applicability,' at Clinton Power Station." In short, the NRC staff concluded that TSSR 3.0.3 was not applicable to the missed inservice testing of the excess flow check valves discussed above because the valves were not tested in accordance with TS required surveillances. The statement of applicability of TSSR 3.0.3 in TS 5.5.6, Inservice Test Program, is only to maintain the allowances for surveillance frequency extensions for TS required surveillances.

The "requirement to declare the LCO not met" referred to in TSSR 3.0.3 is contained in TSSR 3.0.1 and the allowance to delay compliance with that requirement is available for TSSRs that have frequencies governed by the Inservice Testing Program (e.g., TSSR 3.4.4.1). Invoking TSSR 3.0.3 merely allows delaying the declaration that an LCO is not met and does not modify the separate requirement of the LCO that the associated systems be operable. Failure to perform a surveillance of this nature is also a noncompliance and should be resolved as discussed in Regulatory Issue Summary (RIS) 2005-20, "Revision to NRC Inspection Manual Part 9900 Technical Guidance, 'Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety," Revision 1.

Invoking TSSR 3.0.3 for an inservice test that is not a TSSR would be inappropriate because there has been no surveillance that was not performed within its specified frequency, which is necessary to invoke TSSR 3.0.3, and there is no associated requirement from TSSR 3.0.1 to declare an LCO not met. The noncompliance with the inservice testing requirements in 10 CFR 50.55a(f) should also be resolved as discussed in RIS 2005-20, Revision 1.

The TS definition of OPERABLE does not allow a grace period before a component that is not capable of performing its specified function is declared inoperable. Thus, an assessment of the functionality of the excess flow check valves was required to establish whether the degraded/nonconforming condition warranted starting the TS action time for the supported system. This is consistent with NRC staff's Operability Determination Process guidance in RIS 2005-20 for addressing degraded and nonconforming conditions because these valves perform specified functions described in the UFSAR or other elements of the current licensing basis. As stated in this guidance:

> "There is no explicit time limit for completing a prompt determination. Nevertheless, timeliness is important and should depend on the safety significance of the issue. For example, it may be appropriate to make a prompt operability determination within a few hours for situations involving highly safety significant SSCs [structures, systems, or components]. Prompt determinations can often be done within 24 hours of discovery even if complete information is not available. If more time is needed to gather additional information (such as a vendor analyses or calculations), the licensee can evaluate the risk importance of the additional information to decide whether to prolong the operability determination. TS completion time is one factor that can be used in determining an appropriate time frame within which a prompt determination should be completed."

In the current instance, inoperability of the affected SPMU instrumentation would require declaring the SPMU subsystem inoperable within one hour (LCO 3.3.6.4). Inoperability of one SPMU subsystem would require restoring it to operable status within 7 days or changing to Modes 3 and 4 in 12 and 36 hours, respectively. In this context, the delay in conducting an operability determination from November 18, 2008, until June 24, 2009, would be considered excessive.

The inspectors' original question about applying TSSR 3.0.3 when there are multiple missed surveillance intervals was not addressed in TIA 2010-001.

Analysis

The inspectors determined that the licensee's failure to evaluate the functionality of the excess flow check valves to establish whether the nonconforming condition warranted starting the TS action time for the SPMU system was a licensee performance deficiency warranting a significance evaluation. The inspectors assessed this finding using the SDP. The inspectors reviewed the examples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and found no examples related to this issue. Consistent with the guidance in IMC 0612, Appendix B, "Issue Screening," the inspectors determined that the failure to correctly evaluate a degraded/nonconforming condition potentially affecting the operability of SSCs required to be operable by TS would become a more significant safety concern, if left uncorrected, and was, therefore, more than a minor concern, because it could reasonably result in an unrecognized condition of an SSC failing to fulfill a safety-related function. Because the SPMU system was primarily associated with long term decay heat removal following certain design basis accidents, the inspectors concluded that this issue was associated with the Mitigating Systems Cornerstone. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." In accordance with Table 4a, "Characterization Worksheet for IE [Initiating Events], MS [Mitigating Systems], and BI [Barrier Integrity] Cornerstones," the inspectors determined that that this finding was a licensee performance deficiency of very low safety significance (Green) because the finding: (1) was not a design or qualification deficiency; (2) did not represent an actual loss of safety function of a system; (3) did not represent an actual loss of safety function of a single train for greater than its TS allowed outage time; (4) did not represent an actual loss of safety function of one or more non-TS trains of equipment designated as risk significant; and (5) did not screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event.

Cross-Cutting Aspects

The inspectors concluded that this finding affected the cross-cutting area of human performance. Specifically, the licensee did not have a formal process in place with adequate guidance and training to enable licensed senior reactor operators to properly and promptly evaluate operability in this instance. As a result, senior reactor operators took it for granted that utilizing the relief allowed by TSSR 3.0.3 and performing a risk evaluation obviated the need to address operability of the instrumentation supported by the excess flow check valves for the ASME/ANSI Code noncompliance.
(IMC 0310 H.1(a))

Enforcement

No violation of regulatory requirements was identified. This issue is considered to be a finding. **(FIN 05000461/2010003-03, Operability Assessment of Inservice Testing Surveillance Discrepancies for Excess Flow Check Valves)**. The licensee entered this finding into its corrective action program as AR 01063878.

Unresolved Item 05000461/2009003-04 is closed.

1R19 Post-Maintenance Testing (71111.19)

.1 Post-Maintenance Testing

a. Inspection Scope

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

- WO 00488712-05, "CPS 9053.04 - RHR A B C Valve OP";
- WO 01028895-04, “RHR Pump 'A' Seal Cooler SX Inlet Valve”;
- WO 01082152-02, “Division 2 DG Verify Sync-Check Relay 225-DG1KB”;
- WO 01307425-08, “1GC01PA: High Vibration on GC [Stator Cooling Water] Pump ‘A’”; and
- WO 01321254-03, "Leak Check 1C85D002MG, 1C85D002 Back Up Filter."

The inspectors reviewed the scope of the work performed and evaluated the adequacy of the specified post-maintenance testing. The inspectors verified that the post-maintenance testing was performed in accordance with approved procedures; that the procedures contained clear acceptance criteria, which demonstrated operational readiness and that the acceptance criteria was met; that appropriate test instrumentation was used; that the equipment was returned to its operational status following testing; and, that the test documentation was properly evaluated.

In addition, the inspectors reviewed corrective action program documents associated with post-maintenance testing to verify that identified problems were entered into the licensee's corrective action program with the appropriate characterization. Selected action requests were reviewed to verify that the corrective actions were appropriate and implemented as scheduled.

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

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

.1 Surveillance Testing

a. Inspection Scope

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

- CPS 9015.01, "Standby Liquid Control System Operability." (IST)

The inspectors observed selected portions of the test activity to verify that the testing was accomplished in accordance with plant procedures. The inspectors reviewed the test methodology and documentation to verify that equipment performance was consistent with safety analysis and design basis assumptions, and that testing acceptance criteria were satisfied.

In addition, the inspectors verified that surveillance testing problems were entered into the licensee's corrective action program with the appropriate characterization and significance. Selected action requests were reviewed to verify that corrective actions were appropriate and implemented as scheduled.

This inspection constituted one in-service test inspection sample as defined in IP 71111.22.

b. Findings

No findings of significance were identified.

Cornerstone: Emergency Preparedness

1EP2 Alert and Notification System Evaluation (71114.02)

.1 Alert and Notification System Evaluation

a. Inspection Scope

The inspectors held discussions with Emergency Preparedness (EP) staff regarding the operation, maintenance, and periodic testing of the Alert and Notification System (ANS) in the Clinton Power Station's plume pathway Emergency Planning Zone. The inspectors reviewed monthly trend reports and siren test failure records from March 2008 through March 2010. Information gathered during document reviews and interviews was used to determine whether the ANS equipment was maintained and tested in accordance with Emergency Plan commitments and procedures.

This alert and notification system inspection constituted one inspection sample as defined in IP 71114.02.

b. Findings

No findings of significance were identified.

1EP3 Emergency Response Organization Augmentation Testing (71114.03)

.1 Emergency Response Organization Augmentation Testing

a. Inspection Scope

The inspectors reviewed and discussed with plant EP staff the Emergency Plan commitments and procedures that addressed the primary and alternate methods of initiating an Emergency Response Organization (ERO) activation to augment the on-shift ERO as well as the provisions for maintaining the ERO emergency telephone book. The inspectors also reviewed reports and a sample of corrective action program records of unannounced off-hour augmentation tests, which were conducted from March 2008 through April 2010, to determine the adequacy of post-drill critiques and associated corrective actions. The inspectors reviewed the EP training records of approximately 63 ERO personnel assigned to key and support positions to determine the status of their ERO training.

This emergency response organization augmentation testing inspection constituted one inspection sample as defined in IP 71114.03.

b. Findings

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated NCV of 10 CFR 50.54(q) for the licensee's failure to follow and maintain the Emergency Plan, which meets the standards in 10 CFR 50.47(b) and the requirements in Appendix E to 10 CFR 50. Specifically, the licensee's Emergency Plan calls for the performance of periodic drills to evaluate the ability to augment its ERO. However, the Emergency Plan implementing procedure used for the conduct of these augmentation drills exempts certain ERO members from participation in these drills, a situation which prevents the licensee from fully demonstrating its ability to augment all the ERO positions in a timely manner. The licensee's approved Emergency Plan does not provide for such an exemption.

Description

The Clinton Station Emergency Plan consists of the Exelon Standardized Radiological Emergency Plan and the Clinton Station Annex. The Emergency Plan is supported by a series of Emergency Plan implementing procedures and associated program administrative documents, the intent of which is to provide instructions for the implementation of the program elements identified in the plan. The standardized Emergency Plan states that augmentation drills are used to demonstrate the capability to augment the on-shift staff with a Technical Support Center (TSC), Operations Support Center (OSC), and Emergency Operating Facility (EOF) in a short period after declaration of an emergency. Two methods are described in the Emergency Plan: (1) quarterly unannounced off-hours ERO augmentation drills where no actual travel is required (call-in), and (2) at least once per drill cycle (every 6 years), an off-hours unannounced activation of the ERO with actual response to the emergency facilities (drive-in). The licensee's Emergency Plan specifies a minimum ERO staffing that includes management personnel and skilled/technical personnel. The standardized Emergency Plan implementing procedure for conducting drills and exercises, EP-AA-122-1001, "Drill and Exercise Scheduling, Development and Conduct," Attachment 2, states the call-in augmentation drills are intended to demonstrate the ability to contact "selected" ERO personnel. Mechanical, electrical, and instrument maintenance, radiation protection and chemistry are not required to be contacted for the quarterly call-in augmentation drills. As a result, the call-in quarterly drills do not test all the positions for the 60-minute response augmentation required staffing. Since skilled/technical positions are assigned to the Emergency Plan's minimum staffing requirements for the ERO, an augmentation exercise that exempts selected ERO positions is not an adequate demonstration of the licensee's ability to augment the on-shift staff and activate the emergency response facilities as required by the Emergency Plan and by planning standards 10 CFR 50.47(b)(2) and 10 CFR 50.47(b)(8). Although the skilled/technical personnel would be required to participate in the 6-year drive in drill, the licensee has had difficulty in the past with augmentation during an actual emergency.

Analysis

The use of an implementing procedure that causes the conduct of an activity to be inconsistent with the associated requirements in the licensee's Emergency Plan results in a failure to follow and maintain the Emergency Plan and is a performance deficiency. As a result of the limitations in the procedure, the licensee failed to conduct call-in drills to demonstrate timely augmentation of ERO positions filled by skilled/technical personnel. The deficiency did not impact the NRC's regulatory process or contribute to actual safety consequences; therefore, the performance deficiency was screened using the Emergency Preparedness SDP, as a failure to comply.

The deficiency was determined to be of more than minor significance because it adversely affected the EP cornerstone objective to ensure the licensee is capable of implementing adequate measures to protect the health and safety of the public in a radiological emergency. The failure to conduct the augmentation call-in drills in accordance with the Emergency Plan had the attribute associated with ERO readiness and in the area of ERO augmentation testing. The inspector evaluated the finding using the IMC 0609, Appendix B, Sheet I, "Failure to Comply" Flowchart. The licensee's conduct of augmentation drills failed to comply with an Emergency Plan requirement and was associated with two planning standard requirements for timely augmentation.

Title 10 CFR 50.47(b)(2) required, in part, that timely augmentation of response capabilities be available. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," is used by the NRC to evaluate licensee compliance with the regulations in the absence of an approved alternative. Evaluation criterion II.B.5 required, in part, the licensee be able to augment on-shift capabilities within a short period after declaration of an emergency. Title 10 CFR 50.47(b)(8) required that adequate emergency facilities and equipment are provided and maintained. NUREG-0654 evaluation criterion II.H.4 required that each organization shall provide for timely activation and staffing of the facilities and centers described in the Plan. These requirements are applicable to any individual that the licensee assigns to the ERO and who performs emergency response functions identified in the Emergency Plan.

The inspector evaluated the failure to comply with regulatory requirements and the standards for ERO augmentation as a degraded planning standard function since the licensee's conduct of the augmentation exercises did not include all ERO positions. The finding was determined to be of very low safety significance (Green).

Cross-Cutting Aspects

The inspector determined the finding had a cross-cutting aspect in the problem identification and resolution area with a component in self and independent assessments. The licensee's augmentation call-in drills were not comprehensive to include all ERO augmentation staffing positions. (IMC 0310 P.3(a))

Enforcement

Title 10 CFR 50.54(q) required, in part, the licensee to follow and maintain in effect Emergency Plans which meet the standards in section 50.47(b).

Title 10 CFR 50.47(b)(2), required, in part, timely augmentation of response capabilities is available in key functional areas. The Exelon Standardized Radiological Emergency Plan required, in part, augmentation drills to demonstrate the capability of the process to augment the on-shift staff with a TSC, OSC, and EOF through quarterly unannounced off-hours ERO augmentation call-in drills.

Contrary to the above, the licensee failed to follow and maintain in effect Emergency Plans which meet the standards in § 50.47(b) and the requirements in Appendix E to 10 CFR 50, in that the conduct of augmentation drills did not meet the requirement in the licensee's Emergency Plan because the licensee exempted certain members of the ERO from participation. Because the violation was of very low safety significance and was entered into the licensee's corrective action program as AR 01057263, the violation is being treated as an NCV consistent with Section VI.A.1 of the NRC Enforcement Policy **(NCV 05000461/2010003-04, Inadequate Emergency Plan Augmentation Call-In Drills)**.

1EP4 Emergency Action Level and Emergency Plan Changes (71114.04)

.1 Emergency Action Level and Emergency Plan Changes

a. Inspection Scope

The inspectors conducted a review of all the emergency action level changes and sampled the revisions to the Emergency Plan to evaluate whether the changes identified in the revisions may have decreased the effectiveness of the Emergency Plan. The inspection included a review of the 10 CFR 50.54(q) change process documentation. Since the last NRC Emergency Plan change inspection and in accordance with 10 CFR 50.54(q), the Radiological Emergency Plan Annex for Clinton Station, Revision 15, was implemented based on the licensee's determination that the changes resulted in no decrease in effectiveness of the Emergency Plan and the revised Plan continued to meet the requirements of 10 CFR 50.47(b) and Appendix E to 10 CFR 50. The NRC review of the revisions does not constitute formal approval of the changes; therefore, the emergency action level and Emergency Plan changes remain subject to future NRC inspection in their entirety.

This emergency action level and Emergency Plan changes inspection constituted one inspection sample as defined in IP 71114.04.

b. Findings

No findings of significance were identified.

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies (71114.05)

.1 Correction of Emergency Preparedness Weaknesses and Deficiencies

a. Inspection Scope

The inspectors reviewed a sample of the Nuclear Oversight staff's 2008 and 2009 audits of the Clinton Power Station Emergency Preparedness Program to determine if the independent assessments met the requirements of 10 CFR 50.54(t). The inspectors also reviewed critique reports and samples of corrective action program records associated with the 2009 biennial exercise, as well as various EP drills conducted in 2008 and 2009, in order to determine that the licensee fulfilled the drill commitments and to evaluate the licensee's efforts to identify, track, and resolve concerns. Additionally, the inspectors reviewed a sample of corrective actions related to the EP Program and activities to determine whether corrective actions were completed in accordance with the site's corrective action program.

This correction of emergency preparedness weaknesses and deficiencies inspection constituted one inspection sample as defined in IP 71114.05.

b. Findings

No findings of significance were identified.

1EP6 Drill Evaluation (71114.06)

.1 Emergency Preparedness Drill Observation

a. Inspection Scope

The inspectors evaluated the conduct of a full scale emergency preparedness exercise on May 19, 2010, to identify any weaknesses and deficiencies in classification, notification, and protective action recommendation development activities. This drill was planned to be evaluated and was included in performance indicator data regarding drill and exercise performance. The inspectors observed emergency response operations in the Operations 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's drill critique to compare any inspector-observed weaknesses with those identified by the licensee's staff in order to evaluate the critique and to verify whether the licensee's staff was properly identifying weaknesses and entering them into the corrective action program.

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

b. Findings

No findings of significance were identified.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

This inspection constituted one radiological hazard assessment and exposure control inspection sample as defined in IP 71124.01.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed all licensee performance indicators for the Occupational Exposure Cornerstone for follow-up. The inspectors reviewed the results of Radiation Protection Program audits (e.g., licensee's quality assurance audits or other independent audits). The inspectors reviewed any reports of operational occurrences related to occupational radiation safety since the last inspection. The inspectors reviewed the results of the audit and operational report reviews to gain insights into overall licensee performance.

b. Findings

No findings of significance were identified.

.2 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

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

The inspectors selected three air sample survey records and determined whether the samples were collected and counted in accordance with license procedures. The inspectors observed work in potential airborne areas and evaluated whether the air samples were representative of the breathing air zone. The inspectors evaluated whether continuous air monitors were located in areas with low background to minimize false alarms and were representative of actual work areas. The inspectors determined whether the licensee had a program for monitoring levels of loose surface contamination in areas of the plant with the potential for the contamination to become airborne.

b. Findings

No findings of significance were identified.

.3 Instructions to Workers (02.03)

a. Inspection Scope

The inspectors selected five containers holding nonexempt licensed radioactive materials that may cause unplanned or inadvertent exposure of workers, and assessed whether the containers were labeled and controlled in accordance with 10 CFR 20.1904, “Labeling Containers,” or met the requirements of 10 CFR 20.1905(g).

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

The inspectors determined whether, for selected work activities, the licensee had established a means to inform workers of changes in work area conditions that could significantly impact their occupational dose.

b. Findings

No findings of significance were identified.

.4 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

The inspectors observed locations where the licensee monitors potentially contaminated material leaving the radiologically controlled area (RCA), and inspected the methods used for control, survey, and release from these areas. The inspectors observed the performance of personnel surveying and releasing material for unrestricted use and evaluated whether the work was performed in accordance with plant procedures. The inspectors also reviewed whether the procedures were sufficient to control the spread of contamination and prevent unintended release of radioactive materials from the site. The inspectors assessed whether the radiation monitoring instrumentation had appropriate sensitivity for the type(s) of radiation present.

The inspectors reviewed the licensee’s criteria for the survey and release of potentially contaminated material. The inspectors evaluated whether there was guidance on how to respond to an alarm that indicates the presence of licensed radioactive material.

The inspectors reviewed the licensee’s procedures and records to determine whether the radiation detection instrumentation was used at its typical sensitivity level based on appropriate counting parameters. The inspectors assessed whether or not the licensee has established a de facto “release limit” by altering the instrument’s typical sensitivity through such methods as raising the energy discriminator level or locating the instrument in a high-radiation background area.

The inspectors selected three sealed sources from the licensee’s inventory records and assessed whether the sources were accounted for and verified to be intact (i.e., they were not leaking their radioactive content).

The inspectors determined whether any transactions, since the last inspection, involving nationally tracked sources, required reporting or were reported in accordance with 10 CFR 20.2207.

b. Findings

No findings of significance were identified.

.5 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

The inspectors examined the licensee's physical and programmatic controls for highly activated or contaminated materials (nonfuel) stored within spent fuel and other storage pools. The inspectors assessed whether appropriate controls (i.e., administrative and physical controls) were in place to preclude inadvertent removal of these materials from the pool.

The inspectors inspected the posting and physical controls for selected high radiation areas (HRAs) and very high radiation areas (VHRAs), to verify conformance with the Occupational Exposure Control Effectiveness Performance Indicator.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors evaluated licensee controls for VHRAs and areas with the potential to become a VHRA, and ensured that an individual was not able to gain unauthorized access to VHRAs.

b. Findings

No findings of significance were identified.

.7 Radiation Worker Performance (02.07)

a. Inspection Scope

The inspectors reviewed 10 radiological problem reports since the last inspection that found the cause of the event to be human performance errors. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems. The inspectors discussed with the Radiation Protection Manager any problems with the corrective actions planned or taken.

b. Findings

No findings of significance were identified.

.8 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

The inspectors reviewed 10 radiological problem reports since the last inspection that found the cause of the event to be radiation protection technician error. The inspectors evaluated whether there was an observable pattern traceable to a similar cause. The inspectors assessed whether this perspective matched the corrective action approach taken by the licensee to resolve the reported problems.

b. Findings

No findings of significance were identified.

.9 Problem Identification and Resolution (02.09)

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

Cornerstone: Public Radiation Safety

2RS7 Radiological Environmental Monitoring Program and Radioactive Material Control Program (71124.07)

This inspection constituted one Radiological Environmental Monitoring Program and Radioactive Material Control Program inspection sample as defined in IP 71124.07.

.1 Inspection Planning (02.01)

a. Inspection Scope

The inspectors reviewed the annual radiological environmental operating reports and the results of any licensee assessments since the last inspection, to assess whether the Radiological Environmental Monitoring Program (REMP) was implemented in accordance with the TS and the Offsite Dose Calculation Manual (ODCM). This review included report changes to the ODCM with respect to environmental monitoring, commitments in terms of sampling locations, monitoring and measurement frequencies, land use census, inter-laboratory comparison program, and analysis of data.

The inspectors reviewed the ODCM to identify locations of environmental monitoring stations and the UFSAR for information regarding the environmental monitoring program and meteorological monitoring instrumentation.

The inspectors reviewed quality assurance audit results of the program to assist in choosing inspection “smart samples” and audits and technical evaluations performed on the vendor laboratory program.

The inspectors reviewed the annual effluent release report and the 10 CFR Part 61, “Licensing Requirements for Land Disposal of Radioactive Waste,” report, to determine if the licensee was sampling, as appropriate, for the predominant and dose-causing radionuclides likely to be released in effluents.

b. <u>Findings</u>

No findings of significance were identified.

.2 <u>Site Inspection</u> (02.02)

a. <u>Inspection Scope</u>

The inspectors walked down five of the air sampling stations and five of the thermoluminescent dosimeter (TLD) monitoring stations to determine whether they were located as described in the ODCM and to determine the equipment material condition. The air sampling stations were selected based on operability history and included those located in areas of highest effluent deposition based on historical meteorological conditions (X/Q, D/Q wind sectors). Dosimeter monitoring stations were selected based on the most risk-significant locations (e.g., those that have the highest potential for public dose impact). The inspectors reviewed the calibration and maintenance records of several environmental air samplers including those observed during the walkdowns. The records were reviewed to determine whether the equipment was adequately maintained consistent with the licensee’s procedures. The inspectors determined whether the licensee initiated sampling of other appropriate media upon loss of a required sampling station, if applicable.

The inspectors observed the collection and preparation of two environmental samples from different environmental media (e.g., ground and surface water, milk, vegetation, sediment, and soil) as available to assess whether the environmental sampling was representative of the release pathways as specified in the ODCM and that sampling techniques were in accordance with procedures.

Based on direct observation and review of records, the inspectors evaluated whether the meteorological instruments were operable, calibrated, and maintained in accordance with guidance contained in the UFSAR, NRC Regulatory Guide 1.23, “Meteorological Monitoring Programs for Nuclear Power Plants,” and licensee procedures. The inspectors also evaluated whether the meteorological data readout and recording instruments in the control room and, if applicable, at the tower were operable.

The inspectors assessed whether missed and/or anomalous environmental samples were identified and reported in the annual environmental monitoring report. The inspectors selected four events that involved a missed sample, inoperable sampler, lost TLD, or anomalous measurement to evaluate whether the licensee identified the cause and implemented corrective actions. The inspectors reviewed the licensee's assessment of any positive sample results (i.e., licensed radioactive material detected above the lower limits of detection (LLDs) and reviewed the associated radioactive effluent release data that was the source of the released material.

Inspectors selected three SSCs that involve or could reasonably involve licensed material for which there is a credible mechanism for licensed material to reach ground water, and assessed whether the licensee implemented a sampling and monitoring program sufficient to detect leakage of these SSCs to ground water.

The inspectors reviewed historical records required by 10 CFR 50.75(g) of leaks, spills, and remediation to assess the adequacy of the informational content and its retrievability.

The inspectors reviewed significant changes made by the licensee to the ODCM as the result of changes to the land census, revised deposition calculations and/or changes in assessed meteorological conditions or sampler stations since the last inspection. The inspectors reviewed technical justifications for any changed sampling locations to determine whether the licensee performed the required reviews to ensure that the changes did not affect the licensee's ability to monitor the impacts of radioactive effluent releases on the environment.

The inspectors determined if ODCM required detection sensitivities were met for various sample media (i.e., the samples meet required LLDs). The inspectors reviewed the results of the vendor analytical laboratory quality control program, including the inter-laboratory comparison program, to determine the adequacy of the environmental sample analyses provided by the vendor.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems (02.03)

a. Inspection Scope

The inspectors reviewed various corrective action program documents to determine whether problems associated with the REMP were being identified by the licensee at an appropriate threshold. Additionally, the inspectors determined whether the corrective actions for a selected sample of REMP related problems documented by the licensee were adequately evaluated and resolved.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Review of Submitted Quarterly Data

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the First Quarter 2010 Performance Indicators for any obvious inconsistencies prior to its public release in accordance with IMC 0608, "Performance Indicator Program."

This inspection was not considered to be an inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

.2 Safety System Functional Failures

a. Inspection Scope

The inspectors reviewed a previously identified issue with respect to the identification and reporting of an event under the Safety System Functional Failure Performance Indicator.

This inspection was not considered to be an inspection sample as defined in IP 71151.

b. Findings

In September 2009, the inspectors identified that floor drains in the RHR 'A' Pump Room and the Radwaste Pipe Tunnel were interconnected, which resulted in the plant being in an unanalyzed condition that could have prevented fulfillment of the safety function of the ECCS. The inspectors opened URI 05000461/2009004-01 to review the licensee's evaluation of the condition. This issue is further discussed in Section 1R06.1 of this inspection report.

In its evaluation of the unanalyzed condition, the licensee concluded that it was not reportable to the NRC under the requirement of 10 CFR 50.73(a)(2)(v) as a condition that could have prevented the fulfillment of the safety function of structures or systems that are needed to mitigate the consequences of an accident and, therefore, did not count it as an occurrence under the performance indicator. During this inspection period, the inspectors reviewed the licensee's evaluation and determined that the unanalyzed condition met the 10 CFR 50.73(a)(2)(v)(D) reporting criterion and documented a Severity Level IV non-cited violation of 10 CFR 50.73(a)(1), "Licensee Event Report System," because the licensee had failed to submit a required LER within 60 days after discovery of the condition. Therefore, the licensee should also have counted the event as an occurrence under the Safety System Functional Failure Performance Indicator.

The inspectors noted that the performance indicator was at two occurrences during the Fourth Quarter 2009 and First Quarter 2010, with the Green-to-White threshold at six occurrences. One additional occurrence would not cause the performance indicator to change color. Therefore, the inspectors concluded that the licensee's failure report this additional occurrence under the Safety System Functional Failure Performance Indicator constituted a violation of 10 CFR 50.9, "Completeness and Accuracy of Information," of minor significance and is not subject to enforcement action in accordance with the NRC's Enforcement Policy. The licensee entered this violation into its corrective action program as AR 01080117.

.3 Reactor Coolant System Specific Activity

a. Inspection Scope

The inspectors sampled licensee submittals for the RCS Specific Activity Performance Indicator from the Third Quarter 2009 through the Second Quarter 2010. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's RCS chemistry samples, TS requirements, action requests, event reports, and NRC Integrated Inspection Reports for the period of August 2009 through May 2010, to validate the accuracy of the submittals. The inspectors also reviewed the licensee's corrective action program database to determine if any problems had been identified with the data collected or transmitted for this performance indicator. In addition to record reviews, the inspectors observed a chemistry technician obtain and analyze a RCS sample.

This inspection constituted one RCS Specific Activity Performance Indicator inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

.4 Drill/Exercise Performance

a. Inspection Scope

The inspectors sampled the licensee submittals for the Drill/Exercise Performance Indicator for the Fourth Quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors verified the accuracy of the number of reported drill and exercise opportunities and the licensee's critiques and assessments for timeliness and accuracy of the opportunities. The inspectors reviewed the licensee's documentation for control room simulator training sessions, the 2009 biennial exercise, and other designated drills to validate the accuracy of the submittals.

This inspection constituted one Drill/Exercise Performance Indicator inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

.5 Emergency Response Organization Drill Participation

a. Inspection Scope

The inspectors sampled the licensee submittals for the ERO Drill Participation Performance Indicator for the Fourth Quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the licensee's records and ERO roster to validate the accuracy of the submittals for the number of ERO members assigned to fill key positions and the percentage of ERO members who had participated in a performance enhancing drill or exercise.

This inspection constituted one ERO Drill Participation Performance Indicator inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

.6 Alert and Notification System

a. Inspection Scope

The inspectors sampled the licensee submittals for the Alert and Notification System Performance Indicator for the Fourth Quarter 2009. To determine the accuracy of the performance indicator data reported during those periods, performance indicator definitions and guidance contained in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6, dated October 2009, were used. The inspectors reviewed the records of the licensee's reported number of successful siren operability tests as compared to the number of siren tests conducted during the reporting period to validate the accuracy of the submittals.

This inspection constituted one Alert and Notification System inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection

As discussed in previous sections of this report, the inspectors routinely reviewed issues during baseline inspection activities and plant status reviews to verify that they were being entered into the licensee's corrective action program at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Some minor issues were entered into the licensee's corrective action program as a result of the inspectors' observations; however, they are not discussed in this report.

This inspection was not considered to be an inspection sample as defined in IP 71152.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

The inspectors reviewed repetitive or closely-related issues documented in the licensee's corrective action program to look for trends not previously identified. The inspectors also reviewed action requests regarding licensee-identified potential trends to verify that corrective actions were effective in addressing the trends and implemented in a timely manner commensurate with the significance.

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

b. Assessment and Observations

(1) Overall Effectiveness of Trending Program

The inspectors determined that the licensee's trending program was generally effective at identifying, monitoring, and correcting adverse performance trends. The inspectors reviewed several common cause evaluations performed by the licensee to evaluate potential adverse performance trends. In general, these common cause evaluations were performed well and identified appropriate corrective actions to address adverse trends that were identified. The inspectors did not identify any adverse trends that were not already identified by the licensee and entered into its corrective action program.

(2) Adverse Trend in Human Performance

From the most recent refueling outage, C1R12, the inspectors noted that there were a high number of human performance related prompt investigations initiated by the licensee.

The licensee performed a common cause evaluation, AR 01066830-02, on the subject of human performance/technical human performance for C1R12. In this evaluation, the licensee reviewed 24 different corrective action program products, which contained human performance related issues. In the evaluation, the licensee concluded that the common cause of C1R12 human performance events was inaccurate risk perception by the supplemental workforce and a willingness to proceed in the face of uncertainty. During review of this evaluation, inspectors noted that half of the 24 issues were attributed to supplemental workers and the other half to station personnel. As a corrective action to this issue, the licensee implemented a “Return to Excellence Plan,” focused upon improvements in plant staff performance. This plan has three components: technical human performance, online dose, and work management. The licensee is also taking corrective action to implement technical human performance training in operations and maintenance due to an Exelon fleet-wide initiative to address gaps. Sufficient time has not elapsed for the inspectors to evaluate the effectiveness of these particular corrective actions.

A review of inspection results from the past year revealed five findings documented in quarterly NRC resident inspection reports with human performance cross-cutting aspects. This inspection report documents four. Because there are current corrective actions being implemented and specific human performance issues have been adequately addressed individually to date, this adverse trend in human performance is considered to be an observation at this time. The inspectors have noted that since the refueling outage has completed, there has been a reduction in the number of human performance related incidents.

.3 Annual In-Depth Review Sample

a. Inspection Scope

The inspectors selected the following action request for in-depth review:

- AR 01023530, "Gate Seals Lost During CRVCS [Containment and Reactor Vessel Control System] System Functional Test."

The inspectors verified the following attributes during their review of the licensee's corrective actions for the above action request and other related action requests:

- Complete and accurate identification of the problem in a timely manner, commensurate with its safety significance and ease of discovery;
- Consideration of the extent of condition, generic implications, common cause and previous occurrences;
- Evaluation and disposition of operability/reportability issues;
- Classification and prioritization of the resolution of the problem, commensurate with safety significance;
- Identification of the root and contributing causes of the problem; and
- Identification of corrective actions, which were appropriately focused to correct the problem.

The inspectors discussed the corrective actions and associated action request evaluations with licensee personnel.

This inspection constituted one annual in-depth review inspection sample as defined in IP 71152.

b. Findings and Observations

(1) Failure to Follow Procedure Resulting in 46,500 Gallons of Leakage Into Reactor Cavity

Introduction

A finding of very low safety significance (Green) with an associated NCV of 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," was self-revealed on January 29, 2010, when the dryer cavity gate seal depressurized during the performance of the containment and reactor vessel isolation functional surveillance procedure. When the seal lost pressure, approximately 46,500 gallons of water leaked from the dryer cavity pool into the reactor cavity.

Discussion

During the refueling outage on January 29, 2010, the licensee performed surveillance procedure CPS 9601.04, "Containment Drywell Isolation Auto Actuation." During this procedure an isolation signal is generated, which causes the isolation of some primary containment penetrations, including the service air, to containment. When this occurred, the dryer cavity gate seal should have transferred to the backup air supply bottles, but instead, began to leak water when service air was lost. The total amount of leakage due to this event was approximately 46,500 gallons from the dryer cavity pool into the reactor cavity. In addition, a large amount of this water subsequently leaked past the drywell head, down into the drywell, and onto the drywell floor at the bottom of containment.

The licensee performed Root Cause Report #01023530-16 to determine the cause of the failure of the dryer cavity gate seal to transfer to the backup air supply bottles. During the licensee's evaluation of the event, it was discovered that the proper valve operation sequence had not been followed when the gate seal inflation procedure was performed. Specifically, while performing CPS 8117.11, "Installation and Removal of Upper Containment and Fuel Building Gates," both the "inflate" position valve levers were positioned before the regulator pressures were adjusted, which was contrary to the sequence specified by procedure CPS 8117.11. The Reactor Services maintenance craftsman performing the procedure noted that he was not able to lower the regulator setting below 48 pounds-per-square-inch gage (psig). At that time, he also noted that this value of 48 psig was within the accuracy of the gage for 45 psig, and he, therefore, concluded that it was acceptable. However, because the procedure steps had been performed out of sequence, backpressure from the normal service air header was actually indicated on the backup air bottle pressure regulator gage, and not the air pressure supplied from the backup air bottles. The craftsman stated that he attempted to adjust the regulator pressure, but that he was unable to lower the indicated pressure below 48 psig. The craftsman was interviewed and asked why the inflation valves were opened before the regulators were both set. The answer provided was that the inflation valves were opened first because they were in the cabinet and he would not have to come back to the cabinet after setting the backup supply bottle pressure.

This event resulted in a loss of inventory from the dryer cavity pool and partial flooding of the lower reactor cavity. In addition, due to excess water leakage into the drywell, the drywell drains backed up onto the drywell floor. Increased dose was received by outage personnel to allow time to drain the water and attempt to re-perform cavity decontamination. Additional dose was received for work added for the reassembly of the reactor vessel and removal of the reactor head from the flange to clean and re-inspect the O-rings due to potential wetting from the leakage. An additional concern during this event was the potential to create airborne contamination in containment due to the dryer cavity pool walls being highly contaminated and their potential drying out due to lower pool levels.

The inspectors thoroughly examined the licensee’s root cause evaluation and concluded that the licensee had not neglected any significant issues. The licensee concluded that the root cause was the failure of the Reactor Services maintenance craftsman to follow the correct sequence of procedural steps to properly align the backup air supply bottles for the dryer cavity gate seal. The inspectors concluded that the corrective actions taken by the licensee in response to the documented causes appeared to be appropriate.

Analysis

The inspectors determined that the licensee’s failure to correctly install the upper containment dryer cavity gate in accordance with the licensee’s prescribed procedures was a performance deficiency warranting a significance evaluation. The inspectors assessed this finding using the SDP. The inspectors reviewed the examples of minor issues in IMC 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues,” and determined that Example 4c was related to this issue in that data recorded during installation of the dryer cavity gate seal was incorrect and resulted in backup air bottle supply pressure left outside the acceptable range. In addition, consistent with the guidance in IMC 0612, Appendix B, “Issue Screening,” the inspectors determined that the licensee’s failure to correctly install the upper containment dryer cavity gate could be reasonably viewed as a precursor to a significant event and, if left uncorrected, would potentially lead to a more significant safety concern (i.e., increased dose or personnel contamination). Therefore, the inspectors concluded that this finding was of more than minor safety significance. Because the dryer cavity gate seal is intended to contain highly radioactive fluids within containment, which supports the radiological barrier functions to protect plant workers and the public following serious transients or accidents, the inspectors concluded that this issue was associated with the Barrier Integrity Cornerstone. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609 Appendix G, “Shutdown Operations Significance Determination Process,” Attachment 1, “Significance Determination Process Phase 1 Operational Checklists for Both PWRs and BWRs,” Checklist 6, “BWR Cold Shutdown or Refueling Operation – Time to Boil < 2 Hours: RCS Level < 23’ Above Top of Flange,” and determined that Item II.B.(1), Inventory Control Guidelines, Procedures, was not being met due to the performance deficiency. However, because there was no increase in the likelihood of a loss of RCS inventory and the issue could not result in the loss of RCS level instrumentation, the finding did not require a quantitative Phase 2 or Phase 3 analysis and could, therefore, be screened as very low safety significance (Green).

Cross-Cutting Aspects

The inspectors concluded that this finding affected the cross-cutting area of human performance. Specifically, licensee personnel work practices did not support successful human performance. The licensee's Root Cause Report described the root cause as the maintenance craftsman performed steps out of sequence and failed to comply with the procedure. Therefore, as concluded by the Root Cause Report, in this instance, the licensee did not effectively communicate expectations regarding procedural compliance and, as a result, the Reactor Services maintenance craftsman did not correctly follow the procedure by performing steps out of sequence and restoring a system to service that was incorrectly aligned. (IMC 0310 H.4(b))

Enforcement

Title 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings" required, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Contrary to the above, on January 29, 2010, the licensee failed to accomplish CPS 8117.11, "Installation and Removal of Upper Containment and Fuel Building Pool Gates," Revision 8, in accordance with the procedure. Specifically, the licensee failed to adhere to the correct sequence of procedural steps to properly align the backup air supply bottles for the dryer cavity gate seal. Because of the very low safety significance, this violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy **(NCV 05000461/2010003-05, Failure to Follow Procedure Resulting in Gate Seal Leakage)**. The licensee entered this violation into its corrective action program as AR 00969157.

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

.1 (Closed) LER 05000461/2008-003-00, "Excessive Leakage Through Feedwater Isolation Valve 1B21F032A"

On January 28, 2008, a local leak rate test (LLRT) performed on feedwater primary containment isolation check valve 1B21F032A failed the acceptance criterion in TSSR 3.6.1.3.11. The licensee performed maintenance on the valve and retested it satisfactorily. During the root cause evaluation of an LLRT failure for the opposite train feedwater primary containment isolation check valve (1B21F032B) during the Cycle 12 refueling outage in January 2010, the licensee recognized that an LER was never submitted to report the 1B21F032A test failure two years before. The licensee subsequently reported the 1B21F032A test failure as a condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety in accordance with 10 CFR50.73(b)(3)(ii), and as a condition that could have prevented the fulfillment of the safety function of structures or systems needed to control the release of radioactive material and mitigate the consequences of an accident in accordance with 10 CFR 50.73(a)(2)(v). A licensee-identified NCV of 10 CFR 50.73(a)(1) for the licensee's failure to submit a required LER within 60 days after discovery of the 1B21F032A LLRT failure is discussed Section 4OA7.1 of this inspection report.

During review of the LER, the inspectors identified that the licensee did not report this event as an operation or condition prohibited by the plant's TSs in accordance with 10 CFR 50.73(a)(2)(i)(B). The inspectors reviewed the reporting guidance in NUREG-1022, "Event Reporting Guidelines – 10 CFR 50.72 and 50.73," Revision 2, and noted that an operation or condition prohibited by the TSs existed and is reportable if surveillance testing indicates that equipment was not capable of performing its specified safety function (and was thus inoperable) for a period of time longer than allowed by TSs. NUREG-022 further states that for testing conducted within the required surveillance interval, it should be assumed that the discrepancy occurred at the time of its discovery unless there is firm evidence, based on a review of relevant information such as the equipment history and the cause of failure, to indicate that the discrepancy existed previously. The licensee entered this minor reporting issue into its corrective action program as AR 01076505.

The licensee performed a root cause evaluation following the 1B21F032B valve test failure and determined the cause of the failure of both check valves to satisfy the minimum leakage acceptance criterion was age-related degradation of lubrication on the valve actuator, causing increased friction. Based on the cause determination, the inspectors concluded that this valve had not been capable of performing its specified safety function (and thus was inoperable) for a period of time before its discovery longer than allowed by TS 3.6.1.3. A licensee-identified NCV of TS 3.6.1.3 for operating with 1B21F032A inoperable as a result of the licensee's failure to implement appropriate preventive maintenance is discussed in Section 4OA7.2 of this inspection report. As a corrective action, the licensee implemented a preventive maintenance task to clean and re-lubricate the valve actuator during refueling outages.

Licensee Event Report 05000461/2008-003-00 is closed.

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

.2 <u>(Closed) LER 05000461/2010-002-00, "Excessive Leakage Through Feedwater Isolation Valve 1B21F032B"</u>

On February 3, 2010, after Unit 1 had entered Mode 2 during plant startup from the C1R12 refueling outage, the licensee identified that the LLRT performed on feedwater containment isolation check valve 1B21F032B had failed the acceptance criterion in TSSR 3.6.1.3.11. This was discovered by a plant engineer during review of the LLRT test package. Technical Specification Surveillance Requirement 3.6.1.3.11 required that the combined leakage rate for both primary containment feedwater penetrations be less than or equal to 2.0 gpm. The measured leakage rate for the penetration with 1B21F032B was 2.5 gpm. Operators performing the test on January 19, 2010, and the senior reactor operator reviewing the completed test package upon completion did not recognize that the TSSR acceptance criterion was not met because the acceptance criterion was not stated in the surveillance test procedure. The licensee returned the unit to Mode 4 to perform maintenance on 1B21F032B and to re-test it. The motor-operated feedwater primary containment isolation valve (1B21F065B) in the affected flow path was closed prior to plant startup and had remained closed; therefore, the containment penetration flow path was isolated and no actual leakage path existed.

The licensee reported this valve test failure as a condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety in accordance with 10 CFR50.73(b)(3)(ii), and as a condition that could have prevented the fulfillment of the safety function of structures or systems needed to control the release of radioactive material and mitigate the consequences of an accident in accordance with 10 CFR 50.73(a)(2)(v).

The inspectors identified during review of the LER that the licensee did not report this event as an operation or condition prohibited by the plant's TSs in accordance with 10 CFR 50.73(a)(2)(i)(B). The inspectors reviewed the reporting guidance in NUREG-022, "Event Reporting Guidelines – 10 CFR 50.72 and 50.73," Revision 2, and noted that an operation or condition prohibited by the TSs existed and is reportable if surveillance testing indicates that equipment was not capable of performing its specified safety function (and was thus inoperable) for a period of time longer than allowed by the TSs. NUREG-022 further states that for testing conducted within the required surveillance interval, it should be assumed that the discrepancy occurred at the time of its discovery unless there is firm evidence, based on a review of relevant information such as the equipment history and the cause of failure, to indicate that the discrepancy existed previously. The licensee entered this minor reporting issue into its corrective action program as AR 01076505.

The licensee performed a root cause evaluation and determined the cause for the failure was age-related degradation of lubrication on the valve actuator, causing increased friction. No preventive maintenance had been performed on the valve actuator. It was discovered that the lubrication had dried or hardened, causing the valve actuator to bind and not allow the check valve to fully seat. Based on the cause determination, the inspectors concluded that this valve had not been capable of performing its specified safety function (and thus was inoperable) for a period of time before its discovery longer than allowed by TS 3.6.1.3. A licensee-identified NCV of TS 3.6.1.3 for operating with 1B21F032B inoperable as a result of the licensee's failure to implement appropriate preventive maintenance is discussed in Section 4OA7.3 of this inspection report. As a corrective action, the licensee implemented a preventive maintenance task to clean and re-lubricate the valve actuator during refueling outages.

A licensee-identified NCV of 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings" for the licensee's failure to provide appropriate quantitative or qualitative acceptance criteria in CPS 9861.05D014, "LLRT Data Sheet for 1MC010," Revision 2, is discussed in Section 4OA7.4 of this inspection report.

LER 05000461/2010-002-00 is closed.

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

.3 <u>(Closed) LER 05000461/2010-001-00, "Unanalyzed Leakage Pathway Affecting Residual Heat Removal A Pump Room Flooding Analysis"</u>

During review of plant drawings for floor drain system piping in the ECCS and RCIC Pump Rooms on the 707'0" elevation of the Auxiliary Building, the inspectors identified that floor drains in the RHR 'A' Pump Room appeared to be connected via permanent 4-inch pipe embedded in the floor to floor drains in the Radwaste Pipe Tunnel. The inspectors noted that each of the separate pump rooms was supposedly designed to be isolated from other areas of the plant and not susceptible to flooding from sources external to the pump rooms.

In its evaluation of the unanalyzed condition, the licensee concluded that the condition was not reportable to the NRC under the requirements of 10 CFR 50.73(a)(2)(ii)(B) as a condition that resulted in the nuclear power plant being in an unanalyzed condition that significantly degraded plant safety, and 10 CFR 50.73(a)(2)(v)(D) as a condition that could have prevented the fulfillment of the safety function of structures or systems needed to mitigate the consequences of an accident. The inspectors questioned this conclusion and maintained an Unresolved Item open to review the reporting requirements. The licensee wrote AR 01031977 to address the open questions and subsequently submitted this "voluntary" LER.

The inspectors reviewed LER 05000461/2010-001-00 and did not concur with the licensee's conclusion that the reporting criteria were not met. As discussed in Section 1R06.1.b.(1) of this inspection report, the inspectors determined that the unanalyzed condition met the 10 CFR 50.73(a)(2)(ii)(B) and 50.73(a)(2)(v)(D) reporting criteria and documented a Severity Level IV non-cted violation of 10 CFR 50.73(a)(1), "Licensee Event Report System," because the licensee failed to submit a required LER within 60 days after discovery of a condition that met the above reporting criteria. No immediate corrective actions were taken to address this finding; however, the licensee entered this issue into its corrective action program for evaluation.

Licensee Event Report 05000461/2010-001-00 is closed.

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

.4 <u>Licensee Event Notification 45901, "Invalid Actuation of Division 2 Drywell Ventilation and Drywell Cooling Primary Containment Isolation Valves"</u>

On May 5, 2010, the licensee notified the NRC via telephone that the Division 2 Drywell Ventilation and Drywell Cooling Primary Containment Isolation Valves closed for isolation Groups 11 and 17. The event occurred on March 15, 2010, at 0136 during full power operations. The telephone notification was made in accordance with 10 CFR 50.73(a)(1) within the 60-day reporting requirement.

The inspectors reviewed the event notification, prompt investigation, equipment apparent cause evaluation, and the action request (AR 01042588) documented for the event. The inspectors also reviewed the associated operability evaluation as an inspection sample during the first quarter of 2010. The inspectors concluded that the event was correctly reported in accordance with the requirements. During review of the equipment apparent cause evaluation, the inspectors noted that although the licensee has experienced card failures in the past, the corrective actions for the load driver card failure that caused this event were appropriate and commensurate with the significance of the equipment failure. There was no performance deficiency of significance identified with this issue.

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

4OA5 <u>Other Activities</u>

.1 <u>(Closed) URI 05000461/20100002-07, "Main Turbine Trip During On-Line Testing"</u>

During the performance of turbine on-line testing following the C1R12 refueling outage on February 5, 2010, with Unit 1 at about 17 percent power and the main generator synchronized to the grid, the turbine unexpectedly tripped due to mechanical overspeed. The turbine mechanical trip device was mis-adjusted during the refueling outage, which caused the turbine trip logic to activate prematurely during plant start-up testing. The reactor remained on line while troubleshooting the cause for the turbine trip. The licensee completed repairs and synchronized the unit to the grid on February 8th.

The inspectors opened URI 0500461/2010002-07, pending review of the licensee's cause evaluation of the event, to determine whether there was a performance issue of more than minor significance.

During this inspection period, the inspectors thoroughly reviewed the licensee's root cause evaluation for the turbine trip event and concluded that the licensee had not neglected any likely factors. The root cause was determined to be that Turbine Services personnel had failed to ensure the preferred method of resetting the turbine mechanical trip finger was used prior to measuring the running gap due to less than adequate work instructions. This produced an inaccurate gap check when measured and ultimately resulted in improper gap adjustment.

With the unit at 17 percent power, there was no significant transient on the plant due to the turbine trip. Therefore, the inspectors concluded that the performance issue was of minor significance.

Unresolved Item 0500461/2010002-07 is closed.

.2 (Closed) URI 05000461/2009005-02, "Standby Gas Treatment System Flow/Heater Operability Surveillance Test"

The inspectors determined that the licensee's failure to establish an adequate surveillance test procedure with appropriate quantitative or qualitative acceptance criteria to satisfy the monthly standby gas treatment system surveillance testing requirement in TSSR 3.6.4.3.1 was a finding of very low safety significance (Green). The inspectors documented a finding with an associated NCV of 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings," in NRC Inspection Report 05000461/2010002 (NCV 05000461/2010002-08, Inadequate Test Criteria in Standby Gas Treatment System Flow/Heater Operability Surveillance Test).

Unresolved Item 0500461/2009005-02 is closed.

.3 (Closed) NRC Temporary Instruction 2515/173 Review of the Industry Ground Water Protection Voluntary Initiative

a. Inspection Scope

A NRC assessment was performed of the licensee's implementation of the Nuclear Energy Institute – Ground Water Protection Initiative (NEI-GPI) (dated August 2007 (ML072610036)) at Clinton Power Station (CPS). Under the voluntary initiative, each site was to have developed an effective, technically sound groundwater protection program that aligned with the NEI initiative by August 2008.

The inspectors assessed whether the licensee evaluated work practices that could lead to leaks and spills and performed an evaluation of systems, structures, and components that contain licensed radioactive material to determine potential leak or spill mechanisms.

The inspectors determined if the licensee completed a site characterization of geology and hydrology to identify the predominant ground water gradients and potential pathways for ground water migration from onsite to offsite locations. The inspectors also determined if an onsite ground water monitoring program had been implemented to monitor for potential licensed radioactive leakage into groundwater and that the licensee had provisions for the reporting of its ground water monitoring results.
(See http://www.nrc.gov/reactors/operating/ops-experience/tritium/plant-info.html)

The inspectors reviewed the licensee's procedures for the decision making process for potential remediation of leaks and spills, including consideration of the long-term decommissioning impacts. The inspectors reviewed records of leaks and spills that were recorded in the licensee's decommissioning files to determine if the information was in accordance with 10 CFR 50.75(g).

The inspectors reviewed the licensee's notification protocols to determine whether they were consistent with the Groundwater Protection Initiative and/or State of Illinois statutes. The inspectors assessed whether the licensee identified the appropriate local and state officials and conducted briefings with these officials on its ground water protection initiative. The inspectors also determined whether protocols were established for notification of the applicable local and state officials regarding detection of leaks and spills.

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Resident Inspectors' Exit Meeting

The inspectors presented the inspection results to Mr. M. Kanavos and other members of the licensee's staff at the conclusion of the inspection on July 8, 2010. The licensee acknowledged the findings presented. Proprietary information was examined during this inspection, but is not specifically discussed in this report.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- The results of the Radiological Hazards Assessment and Exposure Control Inspection with Mr. T. Chalmers and other members of the licensee's staff on May 14, 2010. 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.

- The results of the Groundwater Protection Initiative and Radiological Environmental Monitoring Program Inspection with Mr. M. Kanavos and other members of the licensee's staff on June 11, 2010. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

- The results of the Emergency Preparedness Inspection with Mr. M. Kanavos and other members of the licensee's staff on April 16, 2010. The inspectors confirmed that none of the potential report input discussed was considered proprietary. A subsequent exit meeting with Mr. D. Kemper and other members of the licensee's staff by telephone was held on June 30, 2010.

.3 Regulatory Performance Meeting

On June 2, 2010, the NRC held a meeting with the licensee at the Clinton Power Station to discuss the Clinton Power Station annual plant performance assessment.

.4 Public Meeting

On June 2, 2010, the NRC held a public open house meeting at the Clinton Elk's Lodge to engage interested members of the public on the performance of the Clinton Power Station and the role of the NRC in ensuring safe plant operations upon completion of the Clinton Power Station annual plant performance assessment in accordance with Section 09.01 of IMC 0305, "Operating Reactor Assessment Program."

4OA7 Licensee-Identified Violations

The following violations of very low significance (Green and Severity Level IV) were identified by the licensee. The violations met the criteria of Section VI of the NRC Enforcement Policy, for dispositioning as non-cited violations.

.1 Failure to Submit a Required LER

Title 10 CFR 50.73(a)(1) required, in part, that the licensee submit an LER for any event of the type described in this paragraph within 60 days after the discovery of the event. Title 10 CFR 50.73(a)(2)(ii) required, in part, that the licensee report any event or condition that resulted in the nuclear power plant being seriously degraded or in an unanalyzed condition that significantly degraded plant safety.

Title 10 CFR 50.73(a)(2)(v) required, in part, that the licensee report any event or condition that could have prevented the fulfillment of the safety function of structures or systems needed to control the release of radioactive material and mitigate the consequences of an accident. In addition, 10 CFR 50.73(a)(2)(i)(B) required, in part, that the licensee report any operation or condition prohibited by the plant's TSs.

Contrary to the above, the licensee failed to submit a required LER within 60 days after discovery of an event on January 28, 2008. The event involved the failure of a primary containment local leak rate test performed on feedwater system check valve 1B21-F032A. The inspectors determined that this finding was of more than minor significance because the NRC relies on licensees to identify and report conditions or events meeting the criteria specified in the TS and the regulations in order to perform its regulatory function. This is a Severity Level IV violation consistent with Supplement I, Paragraph D.4, of the NRC Enforcement Policy and is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered this violation into its corrective action program as AR 01047133. The licensee subsequently submitted LER 05000461/2008-003-00 on May 17, 2010. Refer to Section 4OA3.1 of this inspection report for the review and closure of the LER.

.2 <u>Failure to Meet TS 3.6.1.3 for Primary Containment Isolation Valve 1B21F032A (Train A Feedwater Primary Containment Isolation Check Valve)</u>

Technical Specification 3.6.1.3 required, in part, that each primary containment isolation valve be operable in Modes 1, 2, and 3. TS 3.6.1.3, Condition C.1 stated that with one or more penetration flow paths with the leakage rate not within the limit restore the leakage rate to within the limit within 4 hours. Technical Specification 3.6.1.3, Condition E stated that if the required action and associated completion time of Condition C is not met, be in Mode 3 within 12 hours and Mode 4 within 36 hours.

Contrary to the above, primary containment isolation valve 1B21F032A was found with leakage in excess of the limit during testing on January 28, 2008. The licensee determined the cause for the failure was age-related degradation of lubrication on the valve actuator, causing increased friction. Based on the cause determination, the inspectors concluded that this valve had not been capable of performing its specified safety function (and thus was inoperable) for a period of time before its discovery longer than allowed by TS 3.6.1.3. The inspectors determined that this violation was associated with a licensee-identified finding of very low safety significance (i.e., Green) during a Phase 2 SDP review using the guidance in IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," because the as-found leakage from the penetration was significantly less than the 1,000% containment volume per day criterion in Table 6.2, "Phase 2 Risk Significance – Type B Findings at Full Power." The inspectors also noted that multiple barriers exist to a large release through the feedwater system lines. This violation of TS 3.6.1.3 is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered this violation into its corrective action program as AR 01047133. The licensee submitted LER 05000461/2008-003-00 on May 17, 2010, but did not characterize the event as a condition prohibited by the plant's TSs. The licensee entered this minor reporting issue into its corrective action program as AR 01076505. Refer to Section 4OA3.1 of this inspection report for the review and closure of the LER.

.3 <u>Failure to Meet TS 3.6.1.3 for Primary Containment Isolation Valve 1B21F032B (Train B Feedwater Primary Containment Isolation Check Valve)</u>

Technical Specification 3.6.1.3 required, in part, that each primary containment isolation valve be operable in Modes 1, 2, and 3. TS 3.6.1.3, Condition C.1 stated that with one or more penetration flow paths with the leakage rate not within the limit restore the leakage rate to within the limit within 4 hours. Technical Specification 3.6.1.3, Condition E stated that if the required action and associated completion time of Condition C is not met, be in Mode 3 within 12 hours and Mode 4 within 36 hours.

Contrary to the above, primary containment isolation valve 1B21F032B was found with leakage in excess of the limit during testing on January 18, 2010. The licensee determined the cause for the failure was age-related degradation of lubrication on the valve actuator, causing increased friction. Based on the cause determination, the inspectors concluded that this valve had not been capable of performing its specified safety function (and thus was inoperable) for a period of time before its discovery longer than allowed by TS 3.6.1.3. The inspectors determined that this violation was associated with a licensee-identified finding of very low safety significance (i.e., Green) during a Phase 2 SDP review using the guidance in IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," because the as-found leakage from the penetration was significantly less than the 1,000% containment volume per day criterion in Table 6.2, "Phase 2 Risk Significance – Type B Findings at Full Power." The inspectors also noted that multiple barriers exist to a large release through the feedwater system lines. This violation of TS 3.6.1.3 is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered this violation into its corrective action program as AR 01025446. The licensee submitted LER 05000461/2010-002-00 on April 1, 2010, but did not characterize the event as a condition prohibited by the plant's TSs. The licensee entered this minor reporting issue into its corrective action program as AR 01076505. Refer to Section 4OA3.2 of this inspection report for the review and closure of the LER.

.4 Inadequate Acceptance Criteria in Surveillance Test Procedure for LLRT of Feedwater Isolation Check Valves

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

Contrary to the above, on or about January 18, 2010, the licensee failed to provide appropriate quantitative or qualitative acceptance criteria in surveillance test procedure CPS 9861.05D014, "LLRT Data Sheet for 1MC010," Revision 2, to enable operators performing testing and evaluating the results of testing for compliance with the limits in TSSR 3.6.1.3.11, an activity affecting quality, to identify when the testing results exceeded the limits. Specifically, the procedure did not contain the relevant acceptance criteria. The inspectors determined that this violation was associated with a licensee-identified finding of very low safety significance (i.e., Green) during a Phase 2 SDP review using the guidance in IMC 0609, Appendix H, "Containment Integrity Significance Determination Process," because the as-found leakage from the penetration was significantly less than the 1,000% containment volume per day criterion in Table 6.2, "Phase 2 Risk Significance – Type B Findings at Full Power." The inspectors also noted that multiple barriers exist to a large release through the feedwater system lines. This violation is being treated as a NCV consistent with Section VI.A of the NRC Enforcement Policy. The licensee entered this violation into its corrective action program as AR 01025446. Refer to Section 4OA3.2 of this inspection report for the review and closure of the LER.

.5 Failure to Complete an Independent Review of All Program Elements of the Emergency Preparedness Program

The licensee identified a finding of very low safety significance with an associated NCV of 10 CFR 50.54(t), "Conditions of Licenses," for the failure to complete an independent review of all program elements of the EP Program. The independent assessment did not evaluate and document the adequacy of the interfaces with state and local governments at an interval not to exceed 12 months for all groups. The Nuclear Oversight Emergency Preparedness Program Audit procedure allowed sampling and did not require all state and local governments to be evaluated for adequacy of interfaces. Specifically, Nuclear Oversight's assessment failed to evaluate the adequacy of interface with Macon and Piatt Counties in 2008. All required audits were conducted in 2009 (Dewitt, Macon, McLean, Piatt and the State of Illinois).

The deficiency was screened using the Emergency Preparedness SDP and determined to be of more than minor significance because the finding was associated with the offsite attribute of the EP Cornerstone and affected the objective to ensure the licensee is capable of implementing adequate measures to protect the health and safety of the public in a radiological emergency. The licensee failed to evaluate the working relationship and interfaces between the offsite and onsite emergency response organizations and as a result, failed to conduct an assessment of the overall conduct and effectiveness of the EP Program. The inspector evaluated the finding using the IMC 0609, Appendix B, Sheet I, “Failure to Comply” Flowchart. The audit program was noncompliant with a regulatory requirement not involving an EP planning standard or a risk significant planning standard; therefore, the finding was determined to be of very low safety significance (Green). The licensee entered this violation into its corrective action program as AR 00889346.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

K. Baker, Design Engineering Senior Manager
R. Campbell, RP Technical Specialist
T. Chalmers, Operations Director
J. Cunningham, Security Manager
A. Darelius, EP Manager
J. Domitrovich, Work Management Director
C. Dunn, Shift Operations Superintendent
S. Fatora, Maintenance Director
R. Frantz, Regulatory Assurance
S. Gackstetter, Training Director
G. Hall, Performance Improvement Program Manager
M. Heger, Mechanical/Structural Design Engineering Manager
N. Hightower, Radiological Engineering Manager
M. Kanavos, Plant Manager
F. Kearney, Site Vice President
D. Kemper, Regulatory Assurance Manager
S. Lakebrink, Mechanical Design Engineering
K. Leffel, Operations Support Manager
S. O'Reley, EP Coordinator
J. Peterson, Regulatory Assurance
F. Pournia, Engineering Director
J. Rappeport, Chemistry Manager
S. Soliman, Senior Chemist
J. Stovall, Radiation Protection Manager
J. Ufert, Fire Marshall
C. VanDenburgh, Nuclear Oversight Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000461/2010003-01	NCV	Failure to Satisfy 10 CFR 50.72 and 50.73 Reporting Requirements (Section 1R06.b.(1))
05000461/2010003-02	NCV	Failure to Perform an Adequate 10 CFR 50.59 Evaluation for CPS Procedure 3711.01 (Section 1R13.b.(1))
05000461/2010003-03	FIN	Operability Assessment of Inservice Testing Surveillance Discrepancies for Excess Flow Check Valves (Section 1R15.b.(1))
05000461/2010003-04	NCV	Inadequate Emergency Preparedness Augmentation Call-In Drills (Section 1EP3.1)
05000461/2010003-05	NCV	Failure to Follow Procedure Resulting in Gate Seal Leakage (Section 4OA2.3.b.(1))

Closed

05000461/2009004-01	URI	Interconnecting Floor Drains Between the Residual Heat Removal 'A' Pump Room and Radwaste Pipe Tunnel (Section 1R06.b.(1))
05000461/2010003-01	NCV	Failure to Satisfy 10 CFR 50.72 and 50.73 Reporting Requirements (Section 1R06.b.(1))
05000461/2010002-06	URI	Questions Regarding 10 CFR 50.59 Evaluation for CPS Procedure 3711.01 Involving Operations with the Potential to Drain the Reactor Vessel (Section 1R13.b.(1))
05000461/2010003-02	NCV	Failure to Perform an Adequate 10 CFR 50.59 Evaluation for CPS Procedure 3711.01 (Section 1R13.b.(1))
05000461/2009003-04	URI	Review of Applicability of TSSR 3.0.3 to Multiple Missed Surveillance Intervals for Excess Flow Check Valves (Section 1R15.b.(1))
05000461/2010003-03	FIN	Operability Assessment of Inservice Testing Surveillance Discrepancies for Excess Flow Check Valves (Section 1R15.b.(1))
05000461/2010003-04	NCV	Inadequate Emergency Preparedness Augmentation Call-In Drills (Section 1EP3.1)
05000461/2010003-05	NCV	Failure to Follow Procedure Resulting in Gate Seal Leakage (Section 4OA2.3.b.(1))
05000461/2008-003-00	LER	Excessive Leakage Through Feedwater Isolation Valve 1B21F032A (Section 4OA3.1)
05000461/2010-002-00	LER	Excessive Leakage Through Feedwater Isolation Valve 1B21F032B (Section 4OA3.2)
05000461/2010-001-00	LER	Unanalyzed Leakage Pathway Affecting Residual Heat Removal A Pump Room Flooding Analysis (Section 4OA3.3)
05000461/2010002-07	URI	Main Turbine Trip During On-line Testing (Section 4OA5.1)
05000461/2009005-02	URI	Standby Gas Treatment System Flow/Heater Operability Surveillance Test (Section 4OA5.2)
2515/173	TI	NRC Temporary Instruction 2515/173 Review of the Industry Ground Water Protection Voluntary Initiative (Section 4OA5.3)

Discussed

05000461/2010002 03	NCV	Unanalyzed Condition of Interconnecting Floor Drains Between the RHR 'A' Pump Room and Radwaste Pipe Tunnel (Section 1R06.1.(b))
05000461/2010002-08	NCV	Inadequate Test Criteria in Standby Gas Treatment System Flow/Heater Operability Surveillance Test (Section 4OA5.2)

LIST OF DOCUMENTS REVIEWED

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

<u>1R01 Adverse Weather Protection</u>

- AR 00100707, "Leakage from VT Cooling Coils Following Restoration"
- AR 01052661, "Improperly Secured Items South of Plant"
- AR 01053631, "Severe Weather Off-Normal Entry"
- CPS 1860.02, "Summer Readiness Operation," Revision 0a
- CPS 1860.01C002, "Cold Weather Restoration Checklist," Revision 5b
- CPS 1860.01C003, "Cold Weather Heater and Heat Trace Operability Checklist," Revision 1
- CPS 4302.01, "Tornado/High Winds," Revision 19
- WC-AA-107, "Seasonal Readiness Re-Write," Revision 6
- Plant System Readiness Reviews
- CPS Summer Readiness March Conference Call notes
- Work Order 01225359-01, "Initiate Cold Weather Restorations IAW 1860.01," March 22, 2010
- Work Order 01232603-01, "Return the 0VA03A Cooling Coil to Service," April 19, 2010
- OP-CL-108-107-1001, "Interface between AmerenIP and Clinton Power Station for Switchyard Operations, Maintenance, and Engineering," Revision 8
- AR 00959044, "SY [Switchyard] Issues Discovered During System Manager Walkdown"
- AR 00934821, "Relay System #1 Test Breaker Found in the Off Position"
- AR 01071939, "Utility Pole Leaning Toward 12 KV Bus Work"
- AR 01036598, "1AP02EA: Tasks Required for New RATs for Summer Readiness"
- AR 01034028, "Voltage Transient on 345 KV South Bus at 362.1 KV"
- AR 00467624, "Numerous Control Cables Not Properly Trained in the Panels"
- AR 00469667, "Breaker 4510 Relay (Boulzer 52) 50 PU, LED Illuminated"
- AR 00467634, "Potential for GE CCVTs, GE-CD31 to Fail Prematurely"
- AR 00480385, "CCVTs in the CPS Switchyard Are Prone to Failure"
- AR 00552451, "345 KV Lights in Main Control Room Deenergized"
- AR 00832377, "MPT 'C' Cooling Fan Hi Vibrations"
- AR 01027314, "1AP08EJ Failed to Close During 4160 V Bus 1B Source Shifting"
- AR 00956663, "South Bus Voltage Momentarily Above 362.2 KV"
- AR 00715427, "South Bus Voltage Exceeded 362.2 KV Due to Low Grid Load"
- AR 01038986, "Excessive Fuzzing in Power Lines to New RATs"
- OP-CL-108-107-1002, "Degraded Grid Actions," Revision 2
- CPS 9082.01, "Offsite Source Power Verification," Revision 39b
- Nuclear Plant Operating Agreement for Clinton Power Station
- Second Revised Interconnection Agreement, dated 11/4/03

<u>1R04 Equipment Alignment</u>

- CPS 3312.01, "Residual Heat Removal," Revision 38a
- CPS 3312.01V001, "Residual Heat Removal Valve Lineup," Revision 17
- CPS 3312.01V002, "Residual Heat Removal Instrument Valve Lineup," Revision 9
- AR 01061692, "1TVY006: RHR 'B' Pump Room Fan Did Not Autostart on Temp"
- AR 01061693, "1TVY005: RHR 'B' Pump Room Fan Did Not Autostart on Temp"

- CPS 3506.01, “Diesel Generator and Support Systems,” Revision 34a
- CPS 3506.01V001, “Diesel Generator and Support Systems Valve Lineup,” Revision 13a
- CPS 3506.01V002, “Diesel Generator and Support Systems Instrument Valve Lineup,” Revision 11b
- CPS 3506.01E001, “Diesel Generator and Support Systems Electrical Lineup,” Revision 18a
- CPS 3310.01, “Reactor Core Isolation Cooling (RI),” Revision 27b
- CPS 3310.01V002, “RCIC Instrument Valve Lineup,” Revision 9e
- CPS 3310.01V001, “Reactor Core Isolation Cooling Valve Lineup,” Revision 12e
- CPS 3310.01E001, “Reactor Core Isolation Cooling Electrical Lineup,” Revision 14b
- M05-1079, “P&ID Reactor Core Isolation Cooling (RCIC) (RI),” Sheet 001, Revision AH
- M05-1079, “P&ID Reactor Core Isolation Cooling (RCIC) (RI),” Sheet 002, Revision AJ

1R05 Fire Protection

- Fire Loading Calculation IP-M-0177, Revision 6
- CPS 3822.17, "Emergency Lighting Battery Pack Verification and Testing," Revision 13d
- AR 01060560, "LL System Classified as Maintenance Rule (A)(2) At Risk
- AR 01040525, "EMER Safe S/D Light at Remote S/D PNL Failed 1LL59BP02E”
- AR 01079601, “NRC Observations in FC Heat Exchanger Room”
- Clinton Power Station Updated Final Safety Analysis Report, Appendix E, “Fire Protection Evaluation Report – Clinton Power Station Unit 1,” Revision 11
- Clinton Power Station Updated Final Safety Analysis Report, Appendix F, “Fire Protection Safe Shutdown Analysis – Clinton Power Station Unit 1,” Revision 11
- OP-AA-201-009, “Control of Transient Combustible Material,” Revision 9
- OP-CL-201-009, “Control of Transient Combustible Material,” Revision 1
- Clinton Power Station Updated Final Safety Analysis Report, Appendix E, “Fire Protection Evaluation Report – Clinton Power Station Unit 1,” Revision 11
- Clinton Power Station Updated Final Safety Analysis Report, Appendix F, “Fire Protection Safe Shutdown Analysis – Clinton Power Station Unit 1,” Revision 11
- OP-AA-201-009, “Control of Transient Combustible Material,” Revision 9
- OP-CL-201-009, “Control of Transient Combustible Material,” Revisions 0 and 1
- CPS 1893.04M400, “712 Fuel: Basement Prefire Plan,” Revision 5
- CPS 1893.04M131, “781 Auxiliary (West): Div 2 Containment Electrical Penetrations Prefire Plan,” Revision 5
- AR 01077996, “NRC Identified Combustibles During Inspection”
- AR 01077973, “NRC Identified Deficiencies”

1R06 Flood Protection Measures

- CPS 4304.01, “Flooding,” Revision 4e
- CPS 4411.03, “Injection/Flooding Sources,” Revision 7
- CPS 3208.01, “Cycled/Makeup Condensate,” Revision 12a
- Clinton Power Station Updated Safety Analysis Report, Revision 13
- NRC Information Notice 2009-006, “Construction-Related Experiences with Flood Protection Features,” July 21, 2009
- SL-4576, “Internal Flooding – Safe Shutdown Analysis and INPO SOER No. 85-5 Comparison Evaluation Report” (Sargent & Lundy), January 31, 1990
- A26-1000-01A, “Auxiliary Building Basement Plan Area 1,” Revision AC
- A26-1000-02A, “Auxiliary Building Basement Plan Area 2,” Revision V
- A26-1000-03A, “Auxiliary Building Basement Plan Area 3,” Revision V
- A26-1000-04A, “Auxiliary Building Basement Plan Area 4,” Revision M

- A26-1000-05A, “Auxiliary Building Basement Plan Area 5,” Revision L
- A30-1000-01C, “Control Building Intermediate Floor Plan – Area 1,” Revision F
- AR 00976295, “ECCS Room Floor Drain Piping Connected to the Radwaste Pipe Tunnel”
- AR 01031977, “Questions Regarding IR 976295 Conclusions”
- AR 01039042, “Suppression Pool to ECCS Room Flood Equalization Levels”
- Apparent Cause Evaluation AR 00976295, “ECCS Room Floor Drain Piping Connected to the Radwaste Pipe Tunnel”
- LER 05000461/2010-001-00, “Unanalyzed Leakage Pathway Affecting Residual Heat Removal ‘A’ Pump Room Flooding Analysis,” March 25, 2010
- EC 380335, “Issue Calculation IP-M-0782, ‘Suppression Pool Equalization Levels,” Revision 0
- Calculation IP-M-0782, “Suppression Pool Equalization Levels,” Revision 0

1R07 Heat Sink Performance

- NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment”
- ER-AA-340-1002, “Service Water Heat Exchanger and Component Inspection Guide,” Revision 4
- ER-AA-340, “GL 89-13 Program Implementing Procedure,” Revision 6
- ER-AA-340-1001, “GL 89-13 Program Implementation Instructional Guide,” Revision 7
- CPS 1003.10, “Clinton Power Station (CPS) Program for NRC Generic Letter 89-13,” Revision 6a
- CPS 8130.01, “Heat Exchanger Maintenance/Repairs,” Revision 2
- CPS 9843.02, “Operational Pressure Testing of Class 1, 2, and 3 Systems,” Revision 41a
- AR 01062474, “1VY02C – Anemometer Does Not Meet Requirements”
- AR 01062536, “Change To Chemical Cleaning Impacted Dose Estimate”
- Work Order 01133424-10, “89-13 Cooling Coil Inspection and Cleaning of Cooling Coil 1VY02AB,” April 29, 2010
- Work Order 01133425-10, “89-13 Cooling Coil Inspection and Cleaning of Cooling Coil 1VY02AA,” April 29, 2010
- Work Order 01133424-12, “EP/MM VT2 Maintenance PMT – 9843.02 for 1VY02AB,” March 10, 2010

1R11 Licensed Operator Requalification Program

- AR 01068322, “TQ-JA-150-08 Increases Admin Burden for NARS Form Completion”
- AR 01073222, “NOS ID: Required ANSI Parameters Missed During Transient Test”
- TQ-AA-150, “Operator Training Program,” Revision 4
- TQ-AA-224, “Exelon Nuclear Training – Implementation Phase,” Revision 4
- TQ-AA-306, “Simulator Management,” Revision 1
- OP-AA-102-104, “Pertinent Information Program,” Revision 1

1R12 Maintenance Effectiveness

- ER-AA-310, "Implementation of the Maintenance Rule," Revision 8
- ER-AA-310-1001, "Maintenance Rule - Scoping," Revision 4
- MA-AA-716-210, "Performance Centered Maintenance (PCM) Process," Revision 9
- NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2
- AR 01025101, “1H13P634 Troubleshooting Results on A2 Subloop for 1B33F060A”
- AR 01024342, "A - RR FCV Troubleshooting is Delaying Reactor Startup"

- AR 01039243, "1B33D003A1: RR Subloop A1 Pressurized Light Did Not Come On"
- AR 00869875, "1B33D003A – RR Hydraulic Power Unit A"
- AR 01022991, "1B33D003A1 - RR Hydraulic Power Unit 1A Subloop A1"
- AR 01023016, "1B33D003A2 - RR Hydraulic Power Unit 1A Subloop A1"
- Common Cause Analysis 00736646-02, "C1R11 LRT Failures for Maint. Rule System 97 Need Evaluated"
- WO 01306068-01, "RR A FCV Keeps Going to Lockout"
- ER-AA-310-1001, "Maintenance Rule – Scoping," Revision 4
- Maintenance Rule Failure Report for Reliability Monitoring of System 97 (Containment Isolation/Integrity and Reactor Coolant Pressure Boundary), April 2007 through April 2010
- Maintenance Rule (a)(1) Determination for Reliability Performance Criteria 97-00-1 and 97-00-04 (Local Leak Rate Testing Program Valves), April 14, 2008
- Common Cause Evaluation (AR 00736646-02), "C1R11 LLRT Failures for Maintenance Rule System 97 Need Evaluated," March 12, 2008
- LER 2008-003-00, "Excessive Leakage Through Feedwater Isolation Valve 1B21F032A," May 17, 2010
- LER 2010-002-00, "Excessive Leakage Through Feedwater Isolation Valve 1B21F032B," April 1, 2010
- Root Cause Evaluation (AR 01025446-16), "Late Identification of Feedwater 32B LLRT Failure," March 29, 2010
- CPS 9861.05D014, "LLRT Data Sheet for 1MC010," Revisions 2 and 2a
- AR 01059704, "Evaluation of 1B21F032A Not Forwarded for System 97 Consideration"
- AR 01025446, "1B21-F032B Fails LLRT Not Identified"
- AR 01047133, "Excessive Leakage From 1B21F032A Not Reported in C1R11"
- AR 00736646-02, "C1R11 LLRT Failures for Maintenance Rule System 97 Need Evaluated"
- AR 01049694, "Nuclear Oversight Identified Internal Inspection of Feedwater Checks Not Implemented"

<u>1R13 Maintenance Risk Assessments and Emergent Work Control</u>

- CPS 9051.05, "HPCS Discharge Header Filled and Flow Path Verification," Revision 27c
- Contingency Plan WC-CL-201, "9051.02 HPCS Valve Operability," Revision 0
- CPS 9051.02, "HPCS Valve Operability Test," Revision 39a
- AR 00720355, "Possible Loss of Reactor Coolant Due to Work Package Inadequacy"
- AR 01017904, "Double Blade Guide Removed with Rod Inserted"
- AR 01020181, "NOS ID OPDRV Evaluation Per 3711.01 Not Performed for RR B"
- AR 01029061, "C1R12 LL – Critical Path Delay – DBG Removal w/Rod Inserted"
- AR 01051306, "NRC Concerns with the 50.59 for the new OPDRV Procedure"
- AR 01052493, "1SX083A Check Valve Failed to Seat During 9831.09D004"
- AR 01053125, "NRC Question RE: RHR A HX Availability During 9861.09"
- AR 01063405, "NRC Review of 50.59 for OPDRV Procedure"
- AR 01068302, "Shut Down Cooling Procedure Needs Enhanced"
- WC-AA-101, "On-Line Work Control Process," Revision 16
- CPS 3211.01, "Shutdown Service Water (SX)," Revision 25a
- CPS 9861.09, "Shutdown Service Water Boundary Valve Leak Testing," Revision 0F
- CPS 9861.09, "Leakage Test on Valve 1SX082A," Revision1A
- Peach Bottom Technical Requirements Manual
- Drawing #S27-1004-03A, "Containment Building Floor Framing Plan El. 803'-3"," Revision AG
- Drawing #S27-1200-00A, "Containment Building Section 7-7 Lower," Revision 13
- Plant Operations Review Committee Meeting Minutes, Meeting Number 09-030, December 9, 2009

- Plant Operations Review Committee Meeting Minutes, Meeting Number 10-002, January 10, 2010
- Plant Operations Review Committee Meeting Minutes, Meeting Number 10-009, April 20, 2010
- Nuclear Energy Institute 96-07, “Guidelines For 10CFR 50.59 Implementation,” Revision 1
- NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” December 1991
- NUMARC 93-01, “Industry Guideline For Monitoring The Effectiveness Of Maintenance at Nuclear Power Plants,” Revision 2
- 50.59 Evaluation Number CL-2010-E-001, “CPS Procedure 3711.01,” Revision 0
- 50.59 Evaluation Number CL-2010-E-001, “CPS Procedure 3711.01,” Revision 1
- EC 357294, “Temporary Configuration Change for CPS 8117.04,” Revision 0
- EC 376912, “Evaluate Hole Size for OPDRV,” Revision 1
- OP-AB-117-101, “Operations with the Potential to Drain the Reactor Vessel (OPDRV),” Revision 0
- LaSalle Station LOP-NB-02, “Operations with the Potential to Drain the Reactor Vessel,” Revision 10
- OU-CL-104, “Shutdown Safety Management Program Clinton Power Station,” Revision 4
- CPS 3711.01, “CPS Operations With The Potential To Drain The Reactor Vessel (OPDRV),” Revision 0
- CPS 4006.01, “Loss of Shutdown Cooling,” Revision 4d
- CPS 4011.01, “Reactor Cavity Leakage During Refueling,” Revision 4e
- CPS 4011.02, “Spent Fuel Pool Abnormal Water Level Decrease,” Revision 5f
- CPS 8121.06, “Control Rod Drive Removal and Installation (GE-SLDES III),” Revision 0c
- CPS 8121.06C001, “Control Rod Drive Removal and Installation Checklist (GE-SLDES III),” Revision 0b
- CPS 9093.01, “Control Rod/Drive (CRD) Removal Requirements,” Revision 25b
- CPS 9093.01C002, “Multiple Control Rod/Drive (CRD) Removal Verification Checklist,” Revision 29b
- ER-AA-600, “Risk Management,” Revision 5
- ER-AA-600-1012, “Risk Management Documentation,” Revision 8
- ER-AA-310-1001, “On-Line Risk Management,” Revision 6
- WC-AA-101, “On-Line Work Control Process,” Revision 16
- WC-AA-104, “Integrated Risk Management,” Revision 15

1R15 Operability Evaluations

- AR 01065089, “0AP03E: Small Chips in Two Insulators for the ERAT”
- AR 01065232, “0AP03E: ERAT Core Ground Test Results Lower Than Expected”
- AR 01068918, “Un-Attached Ground Cable in SE Corner of RAT SVC Yard”
- AR 01070053, “ERAT SOW Contingency Plans Not Fully Developed”
- AR 01070098, “Lessons Learned From ERAT SOW”
- AR 01078890, “System Manager Trending ID Potentially Increasing ERAT Core”
- Operability Evaluation # 01065232-07, “0AP03E – Emergency Reserve Auxiliary Transformer”
- Operational and Technical Decision Making #1065232-02, “ERAT Core Ground Testing,”
- May 4, 2010
- Adverse Condition Monitoring and Contingency Plan #1065232-08, “ERAT Core Ground Resistance,” May 5, 2010
- Work Order 00934040, “Doble Test ERAT, Megger ET4 Bus (Cable),” May 26, 2010
- Work Order 01335439, “ERAT Core Ground Test Results Lower Than Expected,” June 11, 2010
- Clinton Power Station Technical Specifications

- Clinton Power Station Updated Final Safety Analysis Report, Revision 11
- NRC Regulatory Issue Summary 2005-20, “Revision to NRC Inspection Manual Part 9900 Technical Guidance, ‘Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality or Safety,’” Revision 1
- Memorandum from G. Shear, (USNRC Region III, Division of Reactor Projects) to T. Blount, (USNRC Office of Nuclear Reactor Regulation, Division of Policy and Rulemaking), Subject: Task Interface Agreement – Evaluation of Application of Technical Specification Surveillance Requirement 3.0.3, “Surveillance Requirement Applicability,” at Clinton Power Station (TIA 2010-001), April 19, 2010
- Memorandum from J. Clifford, (USNRC Region I, Division of Reactor Projects) to T. Blount, (USNRC Office of Nuclear Reactor Regulation, Division of Policy and Rulemaking), Subject: Task Interface Agreement (TIA) – Evaluation of Application of Technical Specification (TS) 4.0.3, “Surveillance Requirement Applicability,” at Pilgrim (TIA 2008-004), December 31, 2008
- CL-SURV-03, “Risk Analysis for Missed Surveillance, Failure to Test Various Excess Flow Check Valves Used for Containment Isolation,” Revision 0
- Operability Evaluation (AR 00846540-08), “IST Surveillance Discrepancies for Excess Flow Check Valves,” Revision 0
- AR 01063878, “NRC TIA on Applicability of TSSR 3.0.3 for IST of Excess Flow Check Valves”
- AR 00846540, “IST Surveillance Discrepancies for Excess Flow Check Valves”
- AR 00943162, “NRC Questions Operability Basis for Operability Evaluation 00846540-08”

1R19 Post-Maintenance Testing

- WO 01028895-01, "IM Overhaul Actuator and Replace Accessories," 1SX029A
- WO 01028895-04, "OPS PMT Perform 9052.01, Verify 1SX029A Strokes Correctly"
- AR 01062042, "Maximum Friction Value Exceeds Acceptance Criteria"
- AR 01063858, “Deficiencies Identified During Testing of 1E12-F037A”
- CPS 9052.01, LPCS/RHR A Pumps & LPCS/RHR A Water Leg Pump Operability"
- WO 01321254-03, "OP PMT Leak Check 1C85D002MG, 1C85D002 Back Up Filter"
- CPS 3105.04, "Steam Bypass and Pressure Regulator (SB)," Revision 13
- WO 00488712-05, "OP CPS 9053.04 - Residual Heat Removal (RHR) A B C Valve OP"
- WO 00488712-01, "Perform Thrust Verification and Clean / Inspect 1E12F037A"
- CPS 9053.04C001, "RHR Loop A Valve Operability," Revision 1d
- CPS 9053.04D001, "RHR Loop A Valve Operability Data Sheet," Revision 43c
- CPS 3506.01, “Diesel Generator and Support System (DG),” Revision 34a
- CPS 9080.02, “Diesel Generator 1B Operability Manual and Quick Start Operability,” Revision 49B
- Work Order 01082152-02, “OP PMT Sync Div II DG to Verify Sync-Check Relay 225-DG1KB,” April 15, 2010
- Work Order 01320453, “9080.02B22 OP DG 1B Operability – Monthly Test,” April 19, 2010
- AR 01069618, “Locked Valve Throttling Per 3211.01V001”
- AR 01070102, “Bolts Were Over-torqued During ERAT [Emergency Reserve Auxiliary Transformer] Restoration”
- AR 01057592, “1GC01PA Is Running Hotter Than Normal”
- WO 01307425, “1GC01PA: High Vibration on GC Pump ‘A’”

1R22 Surveillance Testing

- AR 00919953, “Configuration Error Traps in CPS 9915.01 - Enhancement”
- AR 01064429, “Evaluate LLRT on 1MC116 for Stop-It-Now”

- AR 01069981, “IR 00919953 Closed Without Proper Actions/Changes - CCP”
- CPS 9015.01, “Standby Liquid Control System Operability,” Revision 39e
- CPS 9015.01D001, “Standby Liquid Control Pump and Valve Data Sheet,” Revision 37
- Work Order 01308881-01, “9015.01E23 OP SLC Valve Operability (1C41-F001A & F001B),” May 10, 2010
- Work Order 01325404-01, “OP SLC-SQUIB Valve Continuity and Flow Path Verification,” May 10, 2010

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- Clinton Nuclear Power Station Off-Site Emergency Plan Alert and Notification Addendum, “Prepared in Response to FEMA-REP-10 Documentation Criteria Requirements,” November 1985
- Clinton Off-Site Siren Test Plan, December 2007
- Siren Monthly Operability Reports, 2nd Quarter 2008 through 1st Quarter 2010
- Siren Daily Operability Reports, 2nd Quarter 2008 through 1st Quarter 2010
- Exelon Semi-Annual Siren Report, January 1, 2009 through June 30, 2009
- Exelon Semi-Annual Siren Report, July 1, 2009 through December 31, 2009
- AR 00823763, “Clinton Alert and Notification System Reached an Outage of 25 Percent”
- AR 00945792, “11 of 44 Offsite Emergency Sirens Lost Power”

1EP3 Emergency Response Organization Augmentation Testing

- EP-AA-1000, Exelon Standardized Emergency Plan; Revision 20
- EP-AA-1003, “Radiological Emergency Plan Annex for Clinton Station,” Revision 16
- EP-AA-122-1001, “Drill and Exercise Scheduling, Development, and Conduct,” Revision 11
- TQ-AA-113, “ERO Training and Qualification,” Revision 16
- RP-AA-440, “Respiratory Protection Program,” Revision 9
- RP-CL-441, “Evaluation and Selection Process for Radiological Respirator Use,” Revision4
- CPS 7200.32, “Drywell Entries,” Revision 3c
- CPS 1021.01, “Site Communications,” Revision 7a
- CPS 3822.07, “Gaitronics Verification,” Revision 7
- OP-CL-101-102-1001, “CPS Minimum On-Shift Staffing Functions,” Revision 2
- ERO Augmentation Call-in Drill Reports and Detailed Records, March 2008 through April 2010
- Clinton Power Station Emergency Response Organization Duty Team Roster, April 5, 2010
- AR 00812026, “NOS Identified Additional Review of Gaitronics Work Orders Needed”
- AR 01042183, “Degraded Gaitronics Speakers Not Definable in Passport”
- AR 01016904, “Contract Personnel Utilized to Fulfill ERO Positions”
- AR 01046833, “C1R12 LL Contract ERO Radiation Protection Technicians Need Respirator Qualifications”
- AR 01053501, “Outage ERO Respirator Qualifications below Threshold”
- AR 01057419, “NRC Open Item: On-Shift Respirator Qualifications”
- AR 01057263, “NRC NCV Craft Participation in Augmentation Call-In Drills”
- Clinton Station Selected Emergency Response Personnel Training Records
- Clinton Station Off-site Agencies Correspondence and Training Records

1EP4 Emergency Action Level and Emergency Plan Changes

- 10 CFR 50.54(q) Evaluation Package, “Exelon Nuclear Radiological Emergency Plan Annex for Clinton Station,” Revision 15

1EP5 Correction of Emergency Preparedness Weaknesses and Deficiencies

- LS-AA-126-1001, Attachment 2, “FASA Self-Assessment Report Template, Clinton Station 2010 NRC Baseline Program Inspection Readiness Assessment,” February 19, 2010
- NOSA-CPS-09-04 (AR 848185), “Emergency Preparedness Audit for Clinton Station,” April 15, 2009
- NOSA-CPS-08-03 (AR 699076), “Emergency Preparedness Audit for Clinton Power Station,” April 30, 2008
- NOSA-CPS-07-04 (AR 569195), “Emergency Preparedness Audit for Clinton Station,” May 9, 2007
- NOSA-NCS-09-04, “Emergency Preparedness Audit; NOS Objective Evidence Report,” March 30 through April 3, 2009
- NOSA-NCS-08-03; 2008 Emergency Preparedness Audit, “NOS Objective Evidence Report,” March 31 through April 4, 2008
- NOSA-NCS-07-04, “Emergency Preparedness Audit, NOS Objective Evidence Report, Cantera,” May 18, 2007
- NOSA-NCS-07-04, “Emergency Preparedness Audit; NOS Objective Evidence Report, Clinton,” May 2, 2007
- Clinton 2009 NRC Graded Exercise Evaluation Report, “ November 18, 2009
- EP Exercise Objective D.1 Failure; Root Cause Investigation Report Content and Format,” December 11, 2009
- AR 00889346, “Local Counties Were Not Contacted during QDC NOS EP Audit”
- AR 0083421, “High Failure Rate on New Style Pagers”
- AR 00897250, “Training – Ineffective Change Management around EAL Revision”
- AR 00995972, “TSC Failed Facility Objective for I.2 Core Damage”
- AR 00995985, “Failed Demonstration Criteria in TSC and OSC Associated with K.5.4 (Habitability)”
- AR 00778835, “EP Exercise Evaluation for Objective J.6”
- AR 00883223, “Training – EP Performance in Annual License Exams”

1EP6 Drill Evaluation

- AR 01078888, “2010 ERO Exercise OSC Performance Gaps”
- AR 01078881, “2010 ERO Exercise OSC Demonstration Criteria Deficiencies”
- AR 01078879, “2010 ERO Exercise TSC Demonstration Criteria Deficiencies”

2RS1 Radiological Hazard Assessment and Exposure Controls

- AR 918784, “Additional Qualifications Needed For Whole Body Counter,” May 12, 2010
- AR 923634, “Non-Exempt Source Not In Designated Source Locker,”
- AR 924257, “Enhancement to Survey Program,” May 26, 2009
- AR 938734, “Air Sample Not Counted in Accordance with Procedure,” July 4, 2009
- AR 951729, “Scaffold Access Ladder Not Posted Contamination Area,” August 5, 2009
- AR 972937, “Electronic Dosimeter Dose Rate Alarm,” August 30, 2010
- AR 1012322, “Worker Received Electronic Dosimeter Dose Alarm,” January 5, 2010
- AR 1014236, “Secured High Radiation Area Key Detached from Lanyard,” January 8, 2010
- AR 1016247, “TIP Area Accessed Without Proper Notification/Approval,” January 11, 2010
- AR 1016378, “Worker Received Electronic Dosimeter Dose Alarm,” January 14, 2010
- AR 1016424, “TIP Area Access Controls,” January 14, 2010
- AR 1016902, “Worker Received Electronic Dosimeter Dose Alarm,” January 15, 2010

- AR 1018482, “Nuclear Oversight Identified Black Tape Used for Contaminated Boundary,” January 19, 2010
- AR 1018906, “Nuclear Oversight Identified Radiation Area Brief Used Wrong Dose Estimates,” January 20, 2010
- AR 1020025, “Source Closure Device Failed to Properly Retract,” January 22, 2010
- AR 1020960, “Nuclear Oversight Elevation of Inadequate Supervisory Radiation Protection Oversight of Radiation Protection Technologists,” January 25, 2010
- AR 1024118, “Dry Well Entry Prior to Access Control Guard Checks Performed,”
- AR 1024567. “Nuclear Oversight Identification of Improper Posting of a Contamination Area,” February 1, 2010
- AR 1042235, “Sur-Pak Locked High Radiation Area Not Locked,” March 12, 2010
- AR 104309, “Hand-Written Note on Radiation Protection Posting,” March 16, 2010
- AR 1050562, “Radioactive Sources Not Properly Controlled,” March 30, 2010
- AR 1059639, “Procedure Review Identifies Need for Source Report Tracking,” April 21, 2010
- CR 865815, “Senior Leadership Team Not Notified Prior to Exceeding the Dose Estimate for Condensate Polisher Prefilters,” January 12, 2009
- CR 888839, “Higher than Expected Dose Rates on 0WX01TB Tank Top,” March 4, 2009
- CR 876251, “Temporary Shield Package 2009-052 Installed by Wrong Component,” February 4, 2009
- NF-AA-390, “Spent Fuel Pool Material Control,” Revision 4
- RP-AA-350-1001, “Response to Gate House Alarms,” Revision 0
- RP-AA-503, “Unconditional Release Survey Method,” Revision 2
- RP-AA-800, “Control, Inventory and Leak Testing of Radioactive Sources,” Revision 6
- RP-AA-800-001, “Nationally Tracked Source Program,” Revision 0
- ARP-CL-301-101, “Clinton Power Station Radiological Air Sampling Program,” Revision 3
- RP-CL-503-101, “Clinton Power Station Unconditional Release Surveys,” Revision 6
- Check-in Self-Assessment, “Radworker Performance,” July 31, 1009
- Check-in Self-Assessment, “Infield Work Control,” October 8, 2009

2RS7 Radiological Environmental Monitoring Program

- NOSA-CPS-08-04, “Chemistry, Radwaste, Effluent and Environmental Monitoring Audit Report,”; AR 699063, April 16, 2008
- CY-AA-170-1000, “Radiological Environmental Monitoring Program and Meteorological Program Implementation,” Revision 5
- Annual Report on the Meteorological Monitoring Program at the Clinton Power Station 2009, “Murray and Trettel Inc,” March 2, 2010
- Monthly Report on the Meteorological Monitoring Program at the Clinton Power Station, “Murray and Trettel Inc,” February 2010
- Monthly Report on the Meteorological Monitoring Program at the Clinton Power Station, “Murray and Trettel Inc,” March 2010
- Sampling Procedures Manual, “Environmental Incorporated Midwest Laboratory,” Revision 13
- CPS No. 7200.06F00; 10 CFR 50.75(g)(1),”Decommissioning Records of Radiological Spills or Other Unusual Occurrences,” February 3,1999
- RP-AA-228 Attachment 1, “Record for 10 CFR 50.75(g),” April 9, 2010
- NOVA-0980, “Audit No. SR-2009-32,” Exelon Audit of Murray and Trettel, Inc., October 20, 2009
- NOVA-09-43, “Audit SR 2008-039; Exelon Audit of Environmental, Inc., June 22, 2009
- NUPIC Joint audit Number 20110; Teledyne Brown Engineering Enviro,”Enviro Services, December 1, 2008

- Annual Report on the Meteorological Monitoring Program at the Clinton Power Station 2008, “Murray and Trettel Inc,” September 1, 2009
- AR 01059978, “High Spike in Gross Beta in September 2009,” April 22, 2010
- AR 01059845, Surface Water CL-90 Had False Positive I-131 Result,” April 22, 2010
- AR 01062303, “No flow Through PDCM Water Compositor CL-14,” April 28, 2010
- AR 01079023, “Air Pump Calibration for ODCM Samples,” June 9, 2010

4OA1 Performance Indicator Verification

- CPS 3222.10, “Reactor Sample Station.” Revision 11b
- CPS 6721.01, “Reactor Water Radioisotopic Analysis,” Revision 9
- CY-AA-130-3010, “Dose Equivalent Iodine Determination,” Revision 2
- Nuclear Energy Institute 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6
- HU-AA-101, “Human Performance Tools and Verification Practices,” Revision 4
- Performance Indicator Summary, May 7, 2010
- LS-AA-2110, “Monthly Data Elements for NRC Emergency Response Organization Drill Participation Records; 4th Quarter 2009,”
- LS-AA-2120, Attachment 1, “Monthly Data Elements for NRC Drill/Exercise Performance Reports,” 4th Quarter 2009
- LS-AA-2130, Attachment 1, “Monthly Data Elements for NRC Alert and Notification System Reliability,”4th Quarter 2009

4OA2 Identification and Resolution of Problems

- AR 01023530, “Gate Seals Leakage During CRVCS System Functional Test”
- AR 01029350, “NOS ID Gate Seal Leaking Not Evaluated”
- AR 01035399, “Review SOER 85-1 On Gate Seals”
- AR 01043002, “RR Pump B #1 Seal Cavity Temp, T1, has Risen >4 F/day”
- AR 01043146, “Adverse Trend Identified in Main Control Room Deficiencies”
- AR 01049383, “Trend IR for Possible Declining Trend in Abnormal/Off Normal”
- AR 01066830, “Review of Human Performance Actions on Declining Performance”
- AR 01067144, “Step Change in RR Pump A Seal Pressure”
- AR 01076457, “Extent of Condition for IR 01023530”
- Nuclear Network Operating Experience #1023530-39-01, “Gate Seal Leakage During Containment Isolation Valve System Functional Testing”
- LS-AA-125, “Corrective Action Program (CAP) Procedure,” Revision 11
- Root Cause Investigation #1023530-17, “Gate Seals Leakage During Containment Isolation Valve System Functional Test”
- Prompt Investigation #1023530-6, “Gate Seals Leakage During CRVCS System Functional Test”
- Operations 1st Quarter 2010 Coding and Analysis Report
- Maintenance 1st Quarter 2010 Coding and Analysis Report
- Radiation Protection 1st Quarter 2010 Trending and Analysis Report
- Work Management 1st Quarter 2010 Trending and Analysis Report
- Common Cause Analysis #1066830-02, “Human Performance/Technical Human Performance Common Cause Analysis for C1R12”
- NOL-10-002, Letter from Nuclear Oversight Manager to Radiation Protection Manager, “Elevation – Inadequate Supervisory Oversight to Correct Supplemental Radiation Protection Technician Performance,” January 25, 2010
- CPS 3007.01, “Preparation and Recovery from Refueling Operations,” Revision 14E

- CPS 8117.11, “Installation and Removal of Upper Containment and Fuel Building Pool Gates,” Revision 8C
- CPS 8117.11C001, “Installation and Removal of Upper Containment and Fuel Building Pool Gates Checklist,” Revision 8A
- CPS 9061.04, “Containment Drywell Isolation Auto Actuation,” Revision 42A

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

- Licensee Event Notification 45901, “Division 2 Drywell Ventilation and Drywell Cooling Primary Containment Isolation Valves Closed,” May 5, 2010
- Equipment Apparent Cause Evaluation (AR 01042588), “Division 2 VP/WO Isolation”
- PORC Meeting 10-010 Minutes, May 3, 2010
- PORC Meeting 10-011 Minutes, May 4, 2010
- LER 2008-003-00, “Excessive Leakage Through Feedwater Isolation Valve 1B21F032A,” May 17, 2010
- LER 2010-002-00, “Excessive Leakage Through Feedwater Isolation Valve 1B21F032B,” April 1, 2010
- LER 05000461/2010-001-00, “Unanalyzed Leakage Pathway Affecting Residual Heat Removal ‘A’ Pump Room Flooding Analysis,” March 25, 2010
- Root Cause Evaluation (AR 01025446-16), “Late Identification of Feedwater 32B LLRT Failure,” March 29, 2010
- AR 01025446, “1B21-F032B Fails LLRT Not Identified”
- AR 01042588, “1VP04CB: Div 2 VP/WO Isolation – Shunt Trip”
- AR 01047133, “Excessive Leakage From 1B21F032A Not Reported in C1R11”
- AR 01076505, “Reporting Criteria in LERS for FW 32 LLRT Failures”
- CPS 9861.05D014, “LLRT Data Sheet for 1MC010,” Revisions 2 and 2a

4OA5 Other Activities

- Root Cause Investigation Report (AR 01026445), “Turbine Trip During Turbine On Line Testing,” April 13, 2010
- Hydrogeologic Investigation Report, “Fleetwide Assessment; Clinton Power Station, De Witt County, Illinois; September 2006
- Annual Radioactive Effluent Release Report; Clinton Power Station – Docket Number 50-461, January 1, 2009 – December 31, 2009
- Annual Radiological Environmental Operating Report, “Clinton Power Station – Docket Number 50-461,” April 2010
- CY-CL-170-301, “Offsite Dose Calculation Manual (ODCM),” Clinton Power Station – Docket Number 50-461, Revision 22
- CY-AA-170-400, “Radiological Groundwater Protection Program,” Revision 4
- CY-AA-170-4000, “Radiological Groundwater Protection Program Implementation,” Revision 4
- CY-CY-170-4160, “Radioactive Groundwater Protective Program Scheduling and Notification,” Revision 5
- CY-AA-120-300, “Tritium Management Program,” Revision 1
- NEI Peer Assessment Report, “NEI 07-07 NEI Groundwater Protection Initiative,” February 29, 2010
- Groundwater Protection – Data Collection Questionnaire, July 27, 2006
- AR 00950069, “Evaluate Enhancing RGPP Communications to Local Officials,” August 5, 2009
- AR 01079446, “Annual AREOR Report Does Not Have Complete RGPP Elements,” June 11, 2010

LIST OF ACRONYMS USED

AC	Alternating Current
ADAMS	Agencywide Documents and Management System
ANS	Alert and Notification System
ANSI	American National Standards Institute
AR	Action Request
ASME	American Society of Mechanical Engineers
BI	Barrier Integrity
BWR	Boiling Water Reactor
C1R12	Clinton Unit 1 Cycle 12 Refueling Outage
CFR	Code of Federal Regulations
CNO	Chief Nuclear Officer
CPS	Clinton Power Station
CRDM	Control Rod Drive Mechanism
CRVCS	Containment and Reactor Vessel Cooling System
DG	Diesel Generator
ECCS	Emergency Core Cooling System
EOF	Emergency Operations Facility
EOP	Emergency Operating Procedure
EP	Emergency Preparedness
ERAT	Emergency Reserve Auxiliary Transformer
ERDS	Emergency Response Data System
ERO	Emergency Response Organization
FSAR	Final Safety Analysis Report
FIN	Finding
GDC	General Design Criteria
GPM	Gallons-Per-Minute
HRA	High Radiation Area
IE	Initiating Events
IEL	Initiating Events Likelihood
IMC	Inspection Manual Chapter
IP	Inspection Procedure
IST	Inservice Testing
LCO	Limiting Condition for Operation
LER	Licensee Event Report
LLDs	Lower Limits of Detection
LLRT	Local Leak Rate Test
LOI	Loss of Inventory
LPCS	Low Pressure Core Spray
MS	Mitigating Systems
NCV	Non-Cited Violation
NEI-GPI	Nuclear Energy Institute – Ground Water Protection Initiative
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NSPS	Nuclear System Protection System
ODCM	Offsite Dose Calculation Manual
OPDRV	Operations with the Potential to Drain the Reactor Vessel
OSC	Operations Support Center
PARS	Publicly Available Records System
PSIG	Pounds-Per-Square-Inch Gage

POS	Plant Operating State
RCA	Radiologically Controlled Area
RCS	Reactor Coolant System
RCIC	Reactor Core Isolation Cooling
REMP	Radiological Environmental Monitoring Program
RHR	Residual Heat Removal
RIS	Regulatory Issue Summary
SDP	Significance Determination Process
SPMU	Suppression Pool Makeup
SSCs	Structures, Systems, and Components
SX	Shutdown Service Water
TIA	Task Interface Agreement
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
TSC	Technical Support Center
TSO	Transmission System Operator
TSSR	Technical “Specification Surveillance Requirement
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VHRA	Very High Radiation Area
WO	Work Order

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Sincerely,

/RA/

Mark A. Ring, Chief
Branch 1
Division of Reactor Projects

Docket No. 50-461
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Jared Heck
Allan Barker
Carole Ariano
Linda Linn
DRSIII
DRPIII
Patricia Buckley
Tammy Tomczak
ROPreports Resource