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

May 4, 2010

Mr. Charles G. Pardee
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 05000461/2010-002

Dear Mr. Pardee:

On March 31, 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 April 8, 2010, with Mr. F. Kearney 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 five NRC-identified findings of very low safety significance were identified. Five of these findings were determined to involve violations of NRC requirements. 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 consistent with Section VI.A.1 of the NRC Enforcement Policy.

If you contest any Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial to the U.S. Nuclear Regulatory Commission, ATTN.: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Clinton Power Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement to the Regional Administrator, Region III, and the NRC Resident Inspector at Clinton Power Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be made available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (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 05000461/2010-002
w/Attachment: Supplemental Information

cc w/encl: Distribution via ListServe

U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	05000461/2010-002
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station, Unit 1
Location:	Clinton, IL
Dates:	January 1 through March 31, 2010
Inspectors:	B. Kemker, Senior Resident Inspector D. Lords, Resident Inspector J. Cassidy, Senior Health Physicist E. Coffman, Reactor Engineer J. Draper, Reactor Engineer M. Holmberg, Senior Reactor Inspector R. Jickling, Senior Emergency Preparedness Inspector R. Langstaff, Senior Fire Protection Specialist D. Sand, Reactor Engineer A. Scarbeary, Reactor Engineer S. Mischke, Resident Inspector, Illinois Emergency Management Agency B. Metrow, Illinois Emergency Management Agency
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

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SUMMARY OF FINDINGS

IR 05000461/2010-002; 01/01/10 – 03/31/10; Clinton Power Station, Unit 1; Fire Protection, Flood Protection Measures, Inservice Inspection Activities, Maintenance Effectiveness, and Surveillance Testing.

This report covers a three-month period of inspection by the resident inspectors and announced baseline inspections by regional inspectors. Six Green findings, five of which had an associated Non-Cited Violation, 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: Initiating Events

- Green. The inspectors identified a finding of very low safety significance with an associated Non-Cited Violation of the Clinton Power Station Unit 1 Operating License (NPF-62, Section 2.F). The licensee failed to implement the Fire Protection Program in accordance with program requirements by failing to follow approved Fire Protection Program procedures for the control of transient combustible materials. The licensee promptly removed the transient combustible materials found by the inspectors.

  The inspectors concluded that this finding could be reasonably viewed as a precursor to a significant event (i.e., a fire affecting more than one train of safe shutdown equipment). Specifically, the presence of transient combustible materials in a combustible free zone could reasonably result in degradation of the fire protection defense-in-depth elements in place to prevent fires from starting and mitigate the consequences of fires. In addition, based on review of Example 4k in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," the issue would not be considered to be of minor significance because the identified transient combustibles were found in a combustible free zone required for separation of redundant trains. The finding was of very low safety significance because the items found in the combustible free zone would not be considered transient combustibles of significance as defined in IMC 0609, Appendix F, "Fire Protection Significance Determination Process," Attachment 2, "Degradation Rating Guidance Specific to Various Fire Protection Program Elements," and, therefore, the issue was assigned a "low degradation" rating. The inspectors concluded that this finding affected the cross-cutting area of problem identification and resolution. Specifically, the licensee missed an opportunity to identify and remove the transient combustible materials while implementing corrective actions for previous inspector identified findings involving the control of transient combustible materials. (IMC 0305 P.1(a)) (Section 1R05.1.b.(1))

- Green. The inspectors identified a finding of very low safety significance with an associated Non-Cited Violation of the Clinton Power Station Unit 1 Operating License (NPF-62, Section 2.F). The licensee failed to implement the Fire Protection Program in accordance with program requirements by failing to follow approved Fire Protection Program procedures for the control of combustible gas cylinders in the plant. As a corrective action, the licensee promptly removed the combustible gas cylinders found by the inspectors.

  The inspectors concluded that this finding was associated with the Protection Against External Factors attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the fire hazard for the affected area was increased by the uncontrolled presence of the compressed gas cylinders. In addition, based on review of Example 4k in IMC 0612, “Power Reactor Inspection Reports,” Appendix E, “Examples of Minor Issues,” the issue would not be considered to be of minor significance because a credible fire scenario involving the identified transient combustibles could affect equipment important to safety. The finding was determined to be of very low safety significance during a Phase 3 Significance Determination Process review since the delta core damage frequency was determined to be negligible. Because a postulated fire in the area where the combustible gas cylinders were found could affect only one train of safe shutdown equipment, the safe shutdown path was not affected by the finding. The inspectors concluded that this finding affected the cross-cutting area of human performance. Specifically, the licensee did not adequately ensure that supervisory and management oversight of work activities involving contractors supported nuclear safety. (IMC 0310 H.4(c)) (Section 1R05.1.b.(2))

- Green. A finding of very low safety significance was self-revealed from an event that resulted in a Unit 1 reactor scram. The licensee failed to correct a non-conforming condition with inadequate response from the feedwater level control system (FWLCS) that caused an automatic reactor scram on February 10, 2008, following an unexpected loss of a reactor recirculation pump. This resulted in a second reactor scram for the same cause on October 15, 2009, following the unexpected loss of a reactor recirculation pump. Because the FWLCS is not safety-related, no violation of regulatory requirements was identified. The FWLCS response was corrected in January 2010 and proper system response was verified by the licensee upon start up from the January-February 2010 refueling outage.

  The finding was of more than minor significance because this issue was associated with the Equipment Performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. Specifically, inadequate FWLCS response resulted in a reactor scram following the unexpected loss of a reactor recirculation pump. The finding was of very low safety significance because the issue: (1) did not contribute to the likelihood of a primary or secondary system loss-of-coolant-accident initiator, (2) did not contribute to both the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions would not be available, and (3) did not increase the likelihood of a fire or internal/external flooding event. The inspectors did not identify a cross-cutting aspect related to this finding. (Section 1R12.b.(1))

Cornerstone: Barrier Integrity

- Green. The inspectors identified a finding of very low safety significance with an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, “Instructions, Procedures, and Drawings,” for the licensee’s failure to follow procedure instructions and record examination limitations for containment pipe-to-penetration weld 1-MS-B-11. The licensee subsequently documented the failure to record the 1-MS-B-11 limited weld examination in the corrective action program. The licensee planned to submit limited containment pipe-to-penetration weld examinations to the NRC for review and approval.

  The finding was of more than minor significance because, if left uncorrected, the failure to document limited weld examinations could become a more significant safety concern. Absent NRC identification, the licensee would not have submitted limited weld examinations to the NRC for approval. Further, the inspector could not determine if the NRC would approve the limited weld surface examinations without a licensee evaluation for the extent of additional coverage possible with volumetric weld examinations. This finding was of very low safety-significance based on answering “no” to each of the Phase 1 screening questions identified in the Containment Barrier column of Table 4a in Attachment 0609.04, “Phase 1 - Initial Screening and Characterization of Findings.” Specifically, this finding did not represent an actual open pathway in the physical integrity of reactor containment. This finding has a cross-cutting aspect in the area of Human Performance, Resources because the licensee did not provide complete, accurate and up-to-date design documents (weld construction drawing) to the non-destructive examination staff. Specifically, the lack of a weld construction drawing, which included the weld profile, appeared to have contributed to the examination staff’s failure to recognize that they had not completely examined the required weld surfaces. (IMC 0310 H.2(c)) (Section 1R08.1.b.(1))

- Green. The inspectors identified a finding of very low safety significance with an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criteria V, “Instructions, Procedures, and Drawings.” The licensee failed to include appropriate quantitative or qualitative acceptance criteria in its surveillance test procedure for fulfilling the monthly surveillance requirement to demonstrate operability of the standby gas treatment (SGT) system as described in the Technical Specification Bases. As corrective action, the licensee revised the procedure to include acceptance criteria that system flow is normal and that no blockage, fan or motor failure, or excessive vibration is detected.

  The finding was of more than minor significance because it was associated with the Procedure Quality Cornerstone attribute for the Control Room and Auxiliary Building and adversely affected the Barrier Integrity Cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, by not providing appropriate acceptance criteria by which the operability of the SGT system trains could be assessed, the ability of the SGT system to collect and treat the design leakage of radionuclides from the primary containment to the secondary containment during an accident could not be assured. The inspectors did not identify a cross-cutting aspect related to this finding. (Section 1R22.b.(1))

Cornerstone: Mitigating Systems

- Green. The inspectors identified a finding of very low safety significance with an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criteria III, “Design Control,” regarding the licensee’s failure to correctly translate the design basis into the design of the Auxiliary Building floor drain system with appropriate margin. The inspectors identified that floor drains in the Residual Heat Removal (RHR) ‘A’ Pump Room and the Radwaste Pipe Tunnel were interconnected, which resulted in the plant being in an unanalyzed condition that degraded plant safety and could have prevented fulfillment of the safety function of the containment suppression pool. To address the immediate operability concern, the licensee plugged the two floor drains in the Radwaste Pipe Tunnel line to prevent communication with the floor drain system in the RHR ‘A’ Pump Room. An exposed vertical section of the drain line 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 finding was of more than minor significance because it was associated with the Design Control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the as-found configuration of the interconnecting floor drains resulted in the plant being in an unanalyzed condition that could have prevented fulfillment of the safety function of the containment suppression pool. Although the finding would represent a loss of safety function in the event of a postulated accident, it was determined to be of very low safety significance during a Phase 3 Significance Determination Process review because the delta core damage frequency was determined to be negligible since the initiating event frequency for flooding due to an RHR pump suction pipe failure was sufficiently low. Because this condition had existed since initial plant construction, the performance issue did not necessarily reflect current licensee performance and no cross-cutting aspect was identified. (Section 1R06.1.b.(1))

B. Licensee-Identified Violations

No violations of significance were identified.

REPORT DETAILS

Summary of Plant Status

Unit 1 was operating at about 90 percent power at the beginning of the inspection period conducting end-of-cycle coast down operation.

On January 11, 2010, the unit was removed from service to commence the Cycle 12 refueling outage (C1R12). On February 5th, the licensee performed a reactor startup and synchronized the unit to the grid upon completion of the 25-day refueling outage. During the performance of turbine on-line testing with the unit 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. The reactor remained on line while troubleshooting the cause for the turbine trip. Following successful rework and testing of the mechanical overspeed trip mechanism, the licensee re-synchronized the unit to the grid on February 8th. Unit 1 reached full power on February 10th.

On March 10, 2010, the licensee reduced power to about 92 percent to repair a steam leak on moisture separator reheater valve 1B21-F600B. The unit was returned to full power the same day and was operated at full power for the remainder of the inspection period.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns (71111.04Q)

a. Inspection Scope

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

- Spent Fuel Pool Cooling System during refueling outage following core offload;
- High Pressure Core Spray (HPCS) System (single train risk-significant system); and
- RHR Train ‘A’ during maintenance on RHR Train ‘B’.

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.

Additional activities were performed during the HPCS system walkdown that were associated with Temporary Instruction (TI) 2515/177, "Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems." These activities are described in Section 1R04.3 of this inspection report.

This inspection constituted three partial system walkdown inspection samples as defined in Inspection Procedure (IP) 71111.04.

b. Findings

No findings of significance were identified.

.2 Semi-Annual Complete System Walkdown (71111.04S)

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the plant service water system to verify the functional capability of the system. This system was selected because it was considered important with respect to maintaining cooling to plant systems for reliable plant operation and to limit the occurrences of plant transients. The inspectors walked down the system to review mechanical and electrical equipment lineups, electrical power availability, system pressure and temperature indications, component labeling, component lubrication, component and equipment cooling, hangers and supports, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding work orders was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the corrective action program database to ensure that system equipment alignment problems were being identified and appropriately resolved.

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

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

On January 24, 2010, the inspectors conducted a walkdown of the HPCS system in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspectors' independent walkdown (TI 2515/177, Section 04.02.c.3).

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

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

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

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

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 C-1, "Containment Drywell - Elevations 723'-1-3/4", 737'-0", 755'-0", 778'-0";
- Fire Zone A-2d, "Auxiliary Building Personnel Hatch Area - Elevation 737'0”;
- Fire Zone A-2f, "Main Steam and Pipe Tunnel - Elevations 727'-0", 755'-0";
- Fire Zone T-1e, "Heater Bay and Tunnel - Elevations 737'-0", 762'-0", 781'-0";
- Fire Zone T-1g, "Heater Bays - Elevations 762'-0", 781'-0"; and
- Fire Zone T-1d, "Condenser Pit - Elevation 712'-0".

The inspectors verified that transient combustibles and ignition sources were appropriately controlled and assessed the material condition of fire suppression systems, manual fire fighting 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 six quarterly fire protection inspection samples as defined in IP 71111.05AQ.

b. Findings

(1) Failure to Control Transient Combustible Materials in Accordance with Fire Protection Program

(Closed) Unresolved Item (URI) 05000461/2009005-01, "Failure to Control Transient Combustible Materials in Accordance with Fire Protection Program."

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated Non-Cited Violation of the Clinton Power Station Unit 1 Operating License (NPF-62, Condition 2.F). The licensee failed to implement the Fire Protection Program in accordance with program requirements by failing to follow approved Fire Protection Program procedures for the control of transient combustible materials.

Discussion

On September 29, 2009, with Unit 1 operating in Mode 1, the inspectors identified unattended transient combustible items (a plastic container with about 1 quart of Mobil DTE 26 motor oil, a plastic container with about 1 pint of Syn-Air lubricating oil, an empty collapsible plastic container, a plastic bottle half-filled with what appeared to be a soap-bubble and water solution used for leak detection on pipe fittings, two paper towels, and other assorted small debris items) underneath two air receiver tanks in the Division 1 Diesel Generator Ventilation Fan Room on the Diesel Generator Building 762' elevation. The area in which these transient combustible items were found contained highly visible red striped paint on the floor and markings indicating the area to be a "Combustible Free Zone" as described in the Clinton Power Station Fire Protection Evaluation Report (Updated Final Safety Analysis Report (UFSAR), Appendix E) or, alternatively, a "Transient Combustible Free Zone" (TCFZ) as described in OP-AA-201-009, "Control of Transient Combustible Material," Attachment 5, "Clinton – Station Specific Information." As stipulated in Attachment 5 of OP-AA-201-009, the placement of transient combustible materials in these areas without prior approval in the form of a Transient Combustible Permit (TCP) and Plant Barrier Impairment and additional compensatory measures is prohibited in Modes 1, 2, and 3. Neither a TCP nor a Plant Barrier Impairment was approved for these transient combustible items and no compensatory measures had been established. The procedure further stated that the TCFZs at Clinton Power Station are provided for the purpose of separating redundant safe shutdown equipment. According to the Fire Protection Evaluation Report, redundant safe shutdown equipment of concern for the Division 1 Diesel Generator Ventilation Fan Room included equipment to support operation of the Division 1 diesel generator and main power feed cables for the Division 2 diesel generator. Consequently, a fire in the room could result in the loss of power from both the Division 1 and Division 2 diesel generators. Upon discovery, the inspectors promptly notified the licensee and the items were removed. The items discovered were determined to be Class A and Class B materials as defined in OP-AA-201-009. It is unknown when these items were placed underneath the air receivers.

The inspectors reviewed the licensee's condition evaluation of this issue. The licensee concluded that the items had likely been under the air receiver tanks for several years based on the layer of dust covering them, but not until sometime after mid-2001 based on the style of the labels found on the containers. The licensee attributed the cause to the difficulty in finding the items without extra effort to look into areas not normally used for storage of materials. The inspectors previously documented findings during the fourth quarter of 2007, first quarter of 2008, and fourth quarter of 2008 involving the licensee's failure to follow approved Fire Protection Program procedures for the control of transient combustible materials. Those findings were attributed to poor worker behaviors with storing or staging work materials in TCFZs while work was ongoing and inadequate walkdowns of the plant's TCFZs following previously identified issues. The licensee noted in its evaluation that corrective actions for these findings included several walkdowns of the plant's TCFZs to attempt to identify and remove materials as an extent of condition investigation; however, these walkdowns were not sufficient to identify and remove all transient combustible materials since items have been found after each walkdown. The inspectors noted that the unattended transient combustible items were readily visible by dropping to one knee and looking under the air receiver tanks with a flashlight from the readily accessible floor area where they are located. The licensee's immediate corrective action for this issue was to remove the combustible items from the TCFZ. The licensee subsequently changed its UFSAR and plant procedures to allow some "negligible quantities" of combustible materials inadvertently or accidently left within the TCFZs based upon an engineering evaluation.

The inspectors opened URI 05000461/2009005-01 to consult with a regional fire protection specialist to review the licensee's engineering evaluation and UFSAR changes and to determine whether the performance deficiency was of more than minor safety significance. Refer to Section 4OA2.2 of this inspection report for a discussion of the regional fire protection specialist's review of the changes made to the licensee's Fire Protection Program.

Analysis

The inspectors determined that this failure to follow the procedural requirements of the Clinton Power Station's Fire Protection Program was a licensee performance deficiency warranting a significance evaluation. The inspectors assessed this finding using the Significance Determination Process (SDP). The inspectors reviewed the examples of minor issues in IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," and found one example related to this issue. Example 4k described a situation where a licensee had not followed requirements of its Fire Protection Plan with respect to the control of transient combustible materials. In this example, the issue would be considered to be of more than minor significance if the identified transient combustibles were in a combustible free zone required for separation of redundant trains. In addition, consistent with the guidance in IMC 0612, Appendix B, “Issue Screening,” the inspectors determined that this failure to follow Fire Protection Program procedural requirements could be reasonably viewed as a precursor to a significant event (i.e., a fire affecting more than one train of safe shutdown equipment). Specifically, the presence of transient combustible materials in a combustible free zone could reasonably result in degradation of the fire protection defense-in-depth elements in place to prevent fires from starting and mitigate the consequences of fires. This finding was associated with the Initiating Events 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 3b, “SDP Screening Worksheet for Initiating Events, Mitigation Systems, and Barriers Cornerstones,” the inspectors determined that this finding affected the fire protection defense-in-depth strategies involving fire prevention and administrative controls. Therefore, the inspectors performed a review of this finding using the guidance provided in IMC 0609, Appendix F, “Fire Protection Significance Determination Process.” In Step 1.1, the inspectors determined that this issue involved the finding category of “Fire Prevention and Administrative Controls.” In Step 1.2, the inspectors referenced IMC 0609, Appendix F, Attachment 2, “Degradation Rating Guidance Specific to Various Fire Protection Program Elements,” and assigned a “low degradation” rating to this finding involving the licensee’s combustible controls program. The inspectors’ conclusion was based on the fact that the items found in the TCFZ would not be considered transient combustibles of significance. The attachment defines transient combustibles of significance as low flash point liquids (below 200ºF) and self-igniting combustibles (oily rags). The flashpoints of both oils that were found are well above 200ºF. In Step 1.3, the inspectors determined that this finding was a licensee performance deficiency of very low safety significance (Green) because the issue was assigned a “low degradation” rating.

Cross-Cutting Aspects

The inspectors concluded that the primary cause of this finding was related to the cross-cutting area of problem identification and resolution. Specifically, the licensee did not self-identify and correct this issue through its corrective action program prior to the inspectors’ identification of these unattended transient combustible items in the plant. The inspectors concluded that the licensee missed an opportunity to identify and remove the transient combustible materials while implementing corrective actions for previous inspector identified findings. (IMC 0310 P.1(a))

Enforcement

The Clinton Power Station Unit 1 Operating License (NPF-62), Condition 2.F requires, in part, that the licensee implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report as amended, and as approved in the Safety Evaluation Report (NUREG-0853), dated February 1982, and Supplement Numbers 1 through 8.

The Clinton Power Station UFSAR, Appendix E, “Fire Protection Evaluation Report,” Section 4.0, “Compliance with Branch Technical Position (BTP) APCSB 9.5-1, Appendix A, Plants Under Construction and Operating Plants,” contains the overall program requirements of the licensee's Fire Protection Program. Paragraph C.2, "Instructions, Procedures, and Drawings" states, in part, that administrative controls that govern the Fire Protection Program should be prescribed by documented instructions, procedures, or drawings and should be accomplished in accordance with these documents. OP-AA-201-009, "Control of Transient Combustible Material," Revision 9, prescribes the licensee's administrative controls governing the control of transient combustible materials at Clinton Power Station. OP-AA-201-009, Attachment 5, "Clinton Station Specific Information," Step 1 requires, in part, that authorization be obtained from the Fire Marshall or designee in the form of a TCP prior to staging or storing exposed Class A combustibles or any Class B combustible material in a TCFZ when the plant is in Mode 1, 2, or 3.

Contrary to the above, the licensee failed to follow OP-AA-201-009, Attachment 5, Step 1, by not having an authorized TCP for unattended Class A and Class B combustible items (a plastic container with about 1 quart of Mobil DTE 26 motor oil, a plastic container with about 1 pint of Syn-Air lubricating oil, an empty collapsible plastic container, a plastic bottle half-filled with what appeared to be a soap-bubble and water solution used for leak detection on pipe fittings, two paper towels, and other assorted small debris items) that were found by the inspectors underneath two air receiver tanks in the Division 1 Diesel Generator Ventilation Fan Room on the Diesel Generator Building 762' elevation on September 29, 2009. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A, of the NRC Enforcement Policy **(NCV 05000461/2010002-01, Failure to Control Transient Combustible Materials in Accordance with Fire Protection Program)**. The licensee entered this violation into its corrective action program as action request (AR) 00972704. Unresolved Item 05000461/2009005-01 is closed.

(2) Failure to Control Combustible Gas Cylinders in Accordance with Fire Protection Program

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated Non-Cited Violation of the Clinton Power Station Unit 1 Operating License (NPF-62, Condition 2.F). The licensee failed to implement the Fire Protection Program in accordance with program requirements by failing to follow approved Fire Protection Program procedures for the control of combustible gas cylinders in the plant.

Discussion

On December 30, 2009, the inspectors identified two unattended compressed gas cylinders containing flammable gasses (oxygen and propylene) in the Auxiliary Building Personnel Hatch Area 737'0" elevation. The gas cylinders were staged outside the entrance to the Reactor Water Cleanup (RT) 'A' Pump Room for planned maintenance during the upcoming refueling outage on the RT system. A TCP was not in evidence in the immediate vicinity of the gas cylinders. This did not meet the licensee's requirements for controlling and staging combustible gasses inside critical buildings in OP-CL-201-009, "Control of Transient Combustible Material."

Based on review of OP-CL-201-209, the inspectors noted the following:

1. In Step 2.7 a "CRITICAL BUILDING" was defined as any permanent insured station building or structure which meets any one of the following criteria: 1) Is safety related. 2) Is necessary for power production. 3) Contains radioactive process or storage of radioactive material.

2. Attachment 4 listed the Auxiliary Building as a critical (safety-related) building.

3. Step 4.4.2.1 required that a TCP from the Fire Marshal/Designee be obtained prior to staging a combustible/flammable gas cylinder inside a safety-related building. Contrary to this requirement, a TCP was not issued that included the oxygen/propylene gas cylinders.

4. Step 4.4.3.9.B, required that oxygen/acetylene torch sets/hoses shall be brought into critical buildings only for the duration of the work activity (e.g., hot work operation). Contrary to this requirement, the inspectors discovered the oxygen/propylene torch set staged in the safety-related area on December 30th for refueling outage work that was not scheduled to begin until after January 11th. The exact duration the oxygen/propylene torch set was staged in the critical building could not be determined; however, a walkdown of the area by the inspectors on December 21st did not identify the torch set as present.

Upon discovery, the inspectors immediately notified the licensee and the licensee promptly removed the combustible gas cylinders from the area. To address the extent of condition, the licensee performed a walkdown of other plant areas. Two instances of staged equipment with deficiencies on the TCPs were found. The licensee wrote AR 01010601 to identify the cause and appropriate corrective actions for this issue.

The inspectors reviewed the licensee's condition evaluation of this issue. The licensee concluded that while the contractor supervisor involved with the work to be performed had obtained a TCP to stage welding materials, the list of work and materials on the permit was vague and did not specifically list the flammable gas cylinders. The supervisor and workers did not recognize that the licensee had specific requirements for staging flammable gas cylinders (Class B combustibles). In addition, the gas cylinders were stored about 10 feet from the location described on the permit. The permit was with the other materials at the approved staging area, but the gas cylinders were not in this staging area. The inspectors noted that an apparent lack of contractor oversight and training on Clinton Power Station's specific requirements for control of transient combustible materials contributed to the incident. The licensee's evaluation followed the format of a Quick Human Performance Investigation. The licensee did not evaluate the potential safety significance of the performance issue.

The inspectors reviewed the Fire Protection Evaluation Report and noted that equipment important to safety in the area of concern (Fire Zone A-2d) included Division 1 electrical cables and instruments. Some of the affected systems included: shutdown service water (SX), reactor core isolation cooling (RCIC), low pressure core spray (LPCS), and the remote shutdown panel.

On January 6, 2010, the inspectors identified that contracted workers who were collecting trash and laundry with a large fiberglass cart had left the cart unattended on the west end landing of the Control Building 828'0" elevation for about 5 minutes while they went into the Containment Building. The area where the cart was found contained highly visible red-striped paint on the floor and markings indicating the area to be a TCFZ. The inspectors noted that leaving the cart containing transient combustible materials unattended would not meet the licensee's requirements in OP-CL-201-009 for controlling transient combustible materials in TCFZs. Unknown to the workers involved, an engineering change, approved in July 2009, had recently removed the designation of TCFZ from the Control Building stairwells and landings. This change was incorporated in the UFSAR and in OP-CL-201-009; however, the markings designating the area as a TCFZ were not removed and the changes made to the licensee's Fire Protection Program were not yet communicated to plant personnel. Because the area was no longer a TCFZ, the inspectors concluded that no violation occurred. However, since the area was still posted as a TCFZ, this observation would further indicate a weakness in the licensee's oversight of contractors just before the refueling outage concerning adherence to Fire Protection Program requirements. The licensee wrote AR 01013399 to address the inspectors' observation.

<u>Analysis</u>

The inspectors determined that this failure to follow the procedural requirements of the Clinton Power Station's Fire Protection Program 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 one example related to this issue. Example 4k described a situation where a licensee had not followed requirements of its Fire Protection Plan with respect to the control of transient combustible materials. In this example, the issue would be considered to be of more than minor significance if a credible fire scenario involving the identified transient combustibles could affect equipment important to safety. As discussed above, in the event of a fire in the Auxiliary Building Personnel Hatch Area fueled by the compressed gas cylinders, equipment important to safety could be affected. In addition, consistent with the guidance in IMC 0612, Appendix B, "Issue Screening," the inspectors determined that the finding was associated with the Protection Against External Factors attribute of the Initiating Events Cornerstone and affected the cornerstone objective to limit the likelihood of those events that upset plant stability and challenge critical safety functions during shutdown as well as power operations. Specifically, the fire hazard for the affected area was increased by the uncontrolled presence of the compressed gas cylinders.

Phase 1 SDP Review

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 3b, "SDP Screening Worksheet for Initiating Events, Mitigation Systems, and Barriers Cornerstones," the inspectors determined that this finding affected the fire protection defense-in-depth strategies involving fire prevention and administrative controls. Therefore, the inspectors performed a review of this finding using the guidance provided in IMC 0609, Appendix F, "Fire Protection Significance Determination Process." In Step 1.1, the inspectors determined that this issue involved the finding category of "Fire Prevention and Administrative Controls." In Step 1.2, the inspectors referenced IMC 0609, Appendix F, Attachment 2, "Degradation Rating Guidance Specific to Various Fire Protection Program Elements," and assigned a "high degradation" rating to this finding involving the

licensee's combustible controls program. The inspectors' conclusion was based on the fact that, although the SDP does not specifically take into account flammable gases, the flammable gases stored in cylinders under pressure could result in ignition of a fire from existing sources of heat or electrical energy comparable to low flashpoint liquids.
In Step 1.3, the inspectors determined that this finding would require additional qualitative screening because the issue was not assigned a "low degradation" rating and potentially affected more than just the ability to reach and maintain cold shutdown conditions. In Step 1.4, the inspectors selected a duration factor of 0.1 from Table 1.4.1 based on a 3 - 30 day estimate for the gas cylinders in the area and a generic fire frequency of 9E-2/year from Table 1.4.2 based on the location being in the Auxiliary Building (i.e., Reactor Building Boiling Water Reactor (BWR)). This resulted in an initial Phase 1 screening change in core damage frequency (ΔCDF) value of 9E-3/year.
The inspectors determined that this finding would require a Phase 2/3 SDP review because the initial screening ΔCDF was greater than the high degradation value of 1E-6/year in Table 1.4.3 for a finding in the category of Fire Prevention and Administrative Controls.

Phase 3 SDP Review

The Region III Senior Reactor Analyst (SRA) performed a Phase 3 SDP evaluation to determine the significance of the finding. The duration of the condition was not known but was determined to be less than 7 days. A duration factor of 2E-2 was therefore used in the analysis. For Fire Zone A-2d, a low or medium likelihood of fire ignition frequency would normally be assigned for transient combustible fires; however, because the finding category involved administrative controls, the ignition frequency was determined using the high likelihood rating from IMC 0609, Appendix F, Attachment 4, "Fire Ignition Source Mapping Information: Fire Frequency, Counting Instructions, Applicable Fire Severity Characteristics, and Applicable Manual Fire Suppression Curves." The fire frequency for this category was 1.7E-3/year. The licensee's Safe Shutdown Analysis credited "Method 2" for safe shutdown in this area. This method uses the automatic depressurization system with low pressure injection systems to maintain safe shutdown. Using Table 2.1.1 of IMC 0609, Appendix F, the unavailability of the safe shutdown path was assigned a remaining capability rating of 1E-2 because the safe shutdown path was not affected by the finding since a postulated fire in the area would affect only one train of safe shutdown equipment. Using these assumptions, the ΔCDF for the finding was estimated to be less than 1E-6/year, which represents a finding of very low safety significance (Green).

Cross-Cutting Aspects

The inspectors concluded that the primary cause of this finding was related to the cross-cutting area of human performance. Specifically, the licensee did not adequately ensure that supervisory and management oversight of work activities involving contractors supported nuclear safety. (IMC 0310 H.4(c))

Enforcement

The Clinton Power Station Unit 1 Operating License (NPF-62), Condition 2.F requires, in part, that the licensee implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report as amended, and as approved in the Safety Evaluation Report (NUREG-0853), dated February 1982, and Supplement Numbers 1 through 8.

The Clinton Power Station UFSAR, Appendix E, “Fire Protection Evaluation Report,” Section 4.0, “Compliance with BTP APCSB 9.5-1, Appendix A, Plants Under Construction and Operating Plants,” contains the overall program requirements of the licensee’s Fire Protection Program. Paragraph C.2, “Instructions, Procedures, and Drawings” states, in part, that administrative controls that govern the Fire Protection Program should be prescribed by documented instructions, procedures, or drawings and should be accomplished in accordance with these documents. OP-CL-201-009, “Control of Transient Combustible Material,” Revision 0, prescribes the licensee’s administrative controls governing the control of transient combustible materials at Clinton Power Station. Step 4.4.2.1 of OP-CL-201-209 required that a TCP from the Fire Marshal/Designee be obtained prior to staging a combustible/flammable gas cylinder inside a safety-related building. In addition, Step 4.4.3.9.B required that oxygen/acetylene torch sets/hoses shall be brought into critical buildings only for the duration of the work activity (e.g., hot work operation).

Contrary to the above, on or about December 30, 2009, the licensee failed to follow OP-CL-201-009, Step 4.4.2.1, by not having an authorized TCP for two unattended compressed gas cylinders containing flammable gasses (oxygen and propylene) that were found by the inspectors in the Auxiliary Building Personnel Hatch Area 737’0” elevation. In addition, the licensee failed to follow OP-CL-201-009, Step 4.4.3.9.B, by staging a welding torch set with the two compressed gas cylinders in a critical building several days before the work activity was scheduled to occur. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A, of the NRC Enforcement Policy **(NCV 05000461/2010002-02, Failure to Control Combustible Gas Cylinders in Accordance with Fire Protection Program)**. The licensee entered this violation into its corrective action program as AR 01010601.

1R06 Flood Protection Measures (71111.06)

.1 (Open) 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 evaluate the potential consequences of the as-found condition in the event of a design basis accident.

This inspection was not considered to be an inspection sample as defined in IP 71111.06.

b. Findings

(1) Unanalyzed Condition of Interconnecting Floor Drains Between the RHR ‘A’ Pump Room and Radwaste Pipe Tunnel

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criteria III, “Design Control,” regarding the licensee’s failure to correctly translate the design basis into the design of the Auxiliary Building floor drain system with appropriate margin. 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 degraded plant safety and could have prevented fulfillment of the safety function of the containment suppression pool.

Discussion

During review of plant drawings for floor drain system piping in the Emergency Core Cooling System (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” 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. First, if the Radwaste Pipe Tunnel were to flood, would the floodwater in the tunnel communicate with and cause flooding in the RHR ‘A’ Pump Room? This could potentially affect operability of the RHR ‘A’ pump. Second, if the RHR ‘A’ Pump Room were to flood because of a postulated pump suction line break, would the suppression pool flood water escape the RHR ‘A’ Pump Room and flow into the radwaste system via the Radwaste Pipe Tunnel? 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).

Inspection of the Radwaste Pipe Tunnel found that the floor drains in question were not plugged. No logbook entries were found in the Floor Drain Plug Log to indicate that the drains had been plugged at anytime in the past. The vertical piping which connects the floor drains from the Radwaste Pipe Tunnel to the RHR ‘A’ Pump Room travels through the LPCS Pump Room. This pipe was located as mapped on drawing A26-1000-03A at plant coordinates V-124 and the pipe was intact. No historical design change documents were posted against the plant drawings to indicate that the configuration was altered from the original plant design. Original engineering calculation (3C10-0485-001) and the 1990 Flood Analysis did not specifically discuss the potential for flood water entering (or leaving) the RHR 'A' Pump Room via this drain line. 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 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 presented an unanalyzed condition that was not consistent with the licensing basis; safe shutdown would be still 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, 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 (LOOP). 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 for 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.

In performing the evaluation, the licensee recognized that a portion of the RHR 'A' pump suction line from the primary containment wall up to the remote manual containment isolation valve, 1E12-F004A, was non-isolable in the event of a pipe break. The Flood Analysis had taken this into account, but assumed that the flood water would be retained by the pump room. Therefore, in the event of a non-isolable pipe break, the flood level in the room would equalize with the suppression pool above the minimum vent cover level. The equalization level for the RHR 'A' Pump Room was stated in CPS 4304.01, "Flooding," Table 2, "Suppression Pool Leak/ECCS Room Equalization Levels," Revision 4e, as 15'5" (727'5" elevation). However, a flow path for water out of the pump room via the floor drains invalidated this assumption in the Flood Analysis.
This non-isolable portion of the piping system is within the moderate-energy portion of the fluid system such that only through-wall leakage cracking is postulated.

The licensee revisited the existing calculation for the piping system and revised the piping stress analysis to exempt this line from moderate-energy leakage cracking. The inspectors reviewed the revised calculation and identified no issues of concern. The inspectors determined that the licensee appropriately used the methodology provided in Section III of the American Society of Mechanical Engineers (ASME) Code to determine that the piping stresses were low enough to demonstrate that the crack exclusion criteria set forth in NUREG-0800 were met.

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 that would drain the suppression pool below the suppression pool high water level assumed in Section D3.6.4 of the UFSAR. This particular scenario had not been considered by the licensee in the evaluation. This portion of the piping system is also 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.

To better understand the impact of a 206 gpm loss of inventory from the suppression pool, the inspectors explored the postulated sequence of events for the above non-isolable pipe break scenario. Starting at the nominal suppression pool level of 19'4", the licensee estimated that it would take about 1.5 hours to reach the minimum suppression pool level of 19'0" per TS 3.6.2.2. Technical Specification 3.6.2.2 would require the licensee to restore suppression pool level above 19'0" within 2 hours or enter Mode 3 in 12 hours and Mode 4 in 36 hours. However, with a LOOP, the main turbine would trip due to loss of main condenser vacuum with loss of the main circulating water pumps and the reactor would scram. After about 9.5 hours, the flood level in the RHR 'A' Pump Room would be at the Control Building floor drain level (720'6" elevation) and water would then begin to flow from the pump room into the Radwaste Pipe Tunnel via the floor drains. After about 19.5 hours, the suppression pool level would reach the minimum vent cover level of 15'1" (727'1" elevation), before which time operators would be directed by Emergency Operating Procedure (EOP)-6 to dump the upper containment pool inventory to the suppression pool, scram the reactor, and perform the emergency depressurization (blowdown) actions of EOP-3. The relatively slow rate of inventory loss from the suppression pool would likely enable operators to establish shutdown cooling using RHR Train 'B' before this time. However, with a reactor scram and loss of the normal heat sink to the main condenser, suppression pool temperature would be expected to increase with operation of the safety valves and RCIC pump, necessitating the use of RHR Train 'B' instead for suppression pool cooling. Technical Specification 3.5.2 would require 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 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". Immediately following plant cooldown, the RCIC storage tank may also have insufficient inventory remaining to support operability of the HPCS pump. Dumping the upper containment pool to the suppression pool would provide additional time before reaching 12’9” (or 11’0”) for the licensee to possibly locate the two floor drains in the Radwaste Pipe Tunnel and plug them to stop the suppression pool drain down. This would involve operator manual actions not considered or credited in the current licensing bases.

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) or 50.73(a)(2)(ii) 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) or 50.73(a)(2)(v) 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. Unresolved Item 05000461/2009004-01 will remain open pending the resolution of questions posed by the inspectors regarding the licensee’s evaluation of the reporting requirements. The licensee wrote AR 01031977 to address the open questions and subsequently submitted a voluntary Licensee Event Report (LER 05000461/2010-001-00, “Unanalyzed Leakage Pathway Affecting Residual Heat Removal A Pump Room Flooding Analysis”).

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 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, 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. Unresolved Item 05000461/2009004-01 will remain open to review the licensee’s evaluation.

Analysis

The inspectors determined that the licensee's failure to correctly translate the design basis into the design of the Auxiliary Building floor drain system with appropriate margin 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 finding was associated with the Design Control attribute of the Mitigating Systems Cornerstone and affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences (i.e., core damage). Specifically, the as-found configuration of the interconnecting floor drains resulted in the plant being in an unanalyzed condition that degraded plant safety and could have prevented fulfillment of the safety function of the containment suppression pool.

Phase 1 SDP Review

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 this finding would require a Phase 3 SDP review because the finding would represent a loss of safety function in the event of a postulated event. In addition, the finding would screen as potentially risk significant due to an internal flooding event using the criteria in Table 4b, "Seismic, Flooding, and Severe Weather Screening Criteria." Specifically, the inspectors answered "yes" to question #1 because the interconnecting floor drains resulted in the loss of function for the RHR 'A' Pump Room's flood protection compartment to prevent draining the suppression pool below the suppression pool high water level in the event of a pipe break. The inspectors similarly answered "yes" to question #2c because the suppression pool supports the functions of all three divisions of the ECCS.

Phase 3 SDP Review

The Region III SRA performed a phase 3 SDP evaluation of the finding. The SRA performed a bounding evaluation by assuming that a random un-isolated pipe failure would lead to a reactor trip and entry into the EOPs due to lowering water level in the suppression pool. In the worst case postulated failure of the suction pipe, the suppression pool would drain and not be available as a suction source for ECCS equipment or for containment heat removal. In this scenario, no loss of coolant accident (LOCA) or LOOP is postulated and the motor driven feedwater pump and condensate systems would still be available for decay heat removal.

A pipe failure in the suction piping from the primary containment wall up to the remote manual containment isolation valve, 1E12-F004A, would be non-isolable. This section of piping was estimated to be approximately 20 feet. The SRA used the licensee's initiating event frequency for flooding due to an RHR pump suction pipe failure from its internal events Probabilistic Risk Assessment (PRA). The licensee's initiating event frequency was 1.5E-6/year. Other piping failures in the room are potentially isolable and the SRA considered that either operators could fail to isolate the leakage or a random failure of the isolation valve could occur. The section of piping that was potentially isolable was assumed to be approximately 100 feet. Using the licensee's pipe break frequency from its internal flooding PRA results combined with an estimated probability of 2E-2 for the failure to isolate the break results in an initiating event frequency of 1.5E-7/year. A conditional core damage probability (CCDP) for this bounding case was estimated using the Standardized Plant Analysis Risk (SPAR) model for Clinton Power Station, Revision 3.50. A reactor trip combined with the unavailability of the suppression pool was modeled. The calculated CCDP was 1.8E-2. The ΔCDF was estimated by combining the initiating event frequencies of pipe failures with the CCDP and assuming that the baseline risk of the plant (i.e., the risk due to this pipe failure absent any performance deficiency) would be negligible. The estimated ΔCDF was 3E-8/year, which represents a finding of very low safety significance (Green).

<u>Cross-Cutting Aspects</u>

The inspectors concluded that because this condition had existed since initial plant construction, the performance issue did not necessarily reflect current licensee performance and no cross-cutting aspect was identified.

<u>Enforcement</u>

10 CFR 50, Appendix B, Criteria III, "Design Control," requires, in part, that measures be established to assure that the design basis for safety-related functions of structures, systems, and components are correctly translated into specifications, drawings, procedures, and instructions. Section 3.8.4.1.1 of the Clinton Power Station UFSAR stated, in part, 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, in part, 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).

Contrary to the above, due to the as-built condition with interconnecting floor drains between the RHR 'A' Pump Room and the Radwaste Pipe Tunnel, non-isolable flooding into the RHR 'A' Pump Room due to a pipe rupture could have drained the suppression pool below the suppression pool high water level assumed in Section D3.6.4 of the UFSAR. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy **(NCV 05000461/2010002-03, Unanalyzed Condition of Interconnecting Floor Drains Between the RHR 'A' Pump Room and Radwaste Pipe Tunnel)**. The licensee entered this violation into its corrective action program as AR 00976295.

Unresolved Item 05000461/2009004-01 remains open pending the resolution of questions posed by the inspectors regarding the licensee's evaluation of the extent of condition and reporting requirements.

1R08 Inservice Inspection Activities (71111.08)

From January 13 through 22, 2010, the inspectors conducted a review of the implementation of the licensee's Inservice Inspection (ISI) Program for monitoring degradation of the reactor coolant system (RCS), emergency feedwater systems, risk-significant piping and components, and containment systems.

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

.1 Piping Systems ISI

a. Inspection Scope

The inspectors observed the following non-destructive examination (NDE) required by the ASME Section XI Code to evaluate compliance with the ASME Code, Section XI, applicable ASME Code Cases and Section V requirements, and if any indications and defects were detected, to determine if these were dispositioned, in accordance with the ASME Code or an NRC-approved alternative requirement.

- Automated ultrasonic examination (UT) of reactor pressure vessel vertical shell weld RPV-V4C and analysis of UT data recorded for weld RPV-3C;
- Manual UT of residual heat removal system welds 1-RH-14-15-2, 1-RH-14-24, and 1-RH-14-25;
- Magnetic particle examination (MT) of the pipe-to-penetration weld 1-MS-D-10.

The inspectors reviewed the following examination records with relevant/recordable conditions/indications identified by the licensee to determine if acceptance of these indications for continued service was in accordance with the ASME Code Section XI or an NRC-approved alternative.

- Report No. C1-010, Elbow-to-Pipe Weld 1RH-9-13-7;
- Report No. C1-042, Safe End-to-Nozzle Weld N4D-W-1.

The inspectors observed portions of the following pressure boundary welds completed for a risk-significant system to determine if the licensee followed an ASME Code Section IX qualified welding procedure, maintained control of foreign material, and to determine if the welder used qualified weld filler material and base material. The inspectors also reviewed weld radiographs, post-weld heat treatment charts, and the weld data records in the work order, to determine if these records met the ASME Code Sections III and XI.

- Field welds No. 11 and No. 14 fabricated during installation of the reactor water cleanup system piping replacement under Work Order 1259085.

b. Findings

(1) Failure to Recognize Examination Limitations for a Containment Penetration Weld

Introduction

The inspectors identified a finding of very low safety-significance and a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures and Drawings," for the licensee's failure to follow procedure instructions and record examination limitations for containment pipe-to-penetration weld 1-MS-B-11.

Description

On January 20, 2010, during observation of MT on the containment pipe-to-penetration weld 1-MS-D-10, the inspectors identified that the licensee NDE staff did not recognize this examination as a limited examination. The weld was examined from the exterior surface accessible from the main steam tunnel. The interior weld surface was also required to be examined by the ASME Code Section XI (Table IWB-2500-1, Category B-K-1, Item B.10.10), but was not accessible because of the enclosed penetration pipe. This configuration should have been recognized by the licensee's NDE staff as a limitation to completion of the ASME Code Section XI required examination area. The inspector's questions prompted licensee staff to recognize this as a limited exam and document the limitation on the final examination record (50 percent of required Code coverage was achieved). However, a similar containment pipe-to-penetration weld 1-MS-B-11 was examined in April 2002, with the same limitation and the licensee staff failed to document the examination as limited.

The inspectors reviewed the pipe-to-penetration construction drawing 105D5975, which identified the weld configuration profile applicable to welds 1-MS-D-10 and 1-MS-B-11. During the 2010 penetration weld examination, the licensee NDE staff did not have this drawing available for review. Further, the smooth ground condition of the exterior weld surface challenged the ability of the NDE staff to determine the extent of the weld (e.g., he transition from weld toe to the pipe base-metal was not easy to see). If the NDE staff had reviewed drawing 105D5975 that contained the weld construction profile, the process for determining the extent of weld surface subject to examination would have been simplified. Therefore, the lack of a weld construction drawing appeared to have contributed to the failure of the NDE staff to recognize the limited weld surface examination. The licensee documented the failure to record the 1-MS-B-11 weld examination limitation in AR 01020871 and planned to submit the limited containment penetration weld examinations to the NRC for review and approval.

Analysis

The inspectors determined that the licensee's failure to recognize and document the limited examination of the penetration-to-pipe welds was a performance deficiency that impacted the Barrier Integrity Cornerstone.

The inspectors determined that this finding was more than minor because, if left uncorrected, the failure to document limited weld examinations could become a more significant safety concern. Absent NRC identification, the licensee would not have submitted limited weld examinations to the NRC for approval using the ASME Code relief request process identified in 10 CFR 50.55a. Further, the inspector could not determine if the NRC would approve the limited weld surface examinations without a licensee evaluation for the extent of additional coverage possible with volumetric weld examinations.

The inspectors completed a significance determination, in accordance with IMC 0609, "Significance Determination Process," Attachment 0609.04, "Phase 1 - Initial Screening and Characterization of findings," Table 4a for the Barrier Integrity Cornerstone. Based on answering "no" to each of the Phase 1 screening questions identified in the Containment Barrier column of Table 4a, the finding was determined to be of very low safety-significance. Specifically, this finding did not represent an actual open pathway in the physical integrity of reactor containment.

Cross-cutting Aspects

This finding has a cross-cutting aspect in the area of Human Performance, Resources because the licensee did not provide complete, accurate, and up-to-date design documents (weld construction drawing) to the NDE staff. (IMC 0310 H.2(c))

Enforcement

Title 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality be performed in accordance with instructions, procedures, and drawings appropriate to the circumstance. Instructions, procedures or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Step 7.6 of MT-EXLN-100V3 "Procedure for Magnetic Particle Examination (Dry Particle, Color Contrast or Wet Particle Fluorescent)," Revision 0, required that the examiner "Record all examination limitations on an Examination Data Sheet."

Contrary to these requirements, on April 12, 2002, following surface examination of weld 1-MS-B-11, the limitation to this examination (50 percent limited due to one-sided access) was not recorded on the Examination Data Sheet. Failure to follow the procedure instruction and document the examination limitation was a violation of 10 CFR 50, Appendix B, Criterion V. Because this violation was of very low safety significance and was entered into the corrective action program (AR 01020871), this violation is being treated as a Non-Cited Violation **(NCV 05000461/2010002-04, Failure to Recognize Examination Limitations for a Containment Penetration Weld)** consistent with Section VI.A.1 of the NRC Enforcement Policy.

.2 Reactor Pressure Vessel Upper Head Penetration Inspection Activities (Not Applicable)

.3 Boric Acid Corrosion Control (Not Applicable)

.4 Steam Generator Tube Inspection Activities (Not Applicable)

.5 Identification and Resolution of Problems

a. Inspection Scope

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

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

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

b. Findings

No findings 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 February 24, 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)

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):

- Reactor Recirculation Loop ‘B’ Discharge Valve (1B33F067B) Failure to Close;
- Feedwater Level Control System (FWLCS); and
- Nuclear System Protection System.

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 three maintenance effectiveness inspection samples as defined in IP 71111.12.

b. Findings

(1) Failure to Correct Inadequate FWLCS Response Resulted in High Reactor Vessel Water Level (Level 8) Scram

Introduction

The inspectors identified a finding of very low safety significance (Green) associated with a self-revealed event. The licensee failed to correct a non-conforming condition with inadequate response from the FWLCS that caused an automatic reactor scram on February 10, 2008, following an unexpected loss of a reactor recirculation pump. This resulted in a second reactor scram for the same cause on October 15, 2009, following the unexpected loss of a reactor recirculation pump. Because the FWLCS is not safety-related, no violation of regulatory requirements was identified.

Discussion

On October 15, 2009, Unit 1 was manually scrammed following an unexpected trip of the ‘B’ reactor recirculation pump. Loss of the reactor recirculation pump resulted in a plant transient, causing reactor power to decrease and reactor vessel water level to rise. Operators manually scrammed the reactor just before reactor vessel level reached the Level 8 (high level) reactor scram setpoint. After the unit was shut down, the licensee identified that the pump motor had failed due to an internal electrical fault.

The Clinton Power Station UFSAR, Section 15.3.1.2.1.3.1 states that no scram occurs for a trip of one reactor recirculation pump. Furthermore, Section 15.3.1.2.2.1 states that tripping a single recirculation pump requires no protection system or safeguard system operation. This analysis assumes normal functioning plant instrumentation and controls. To summarize the event results from Section 15.3.1.3.3.1, no core thermal limits are exceeded and during the transient reactor vessel level swell is not sufficient to cause a turbine trip and reactor scram. The cause for the reactor scram was determined to be inadequate response of the FWLCS to an anticipated plant transient (i.e., trip of one reactor recirculation pump) that should not have resulted in a scram. This inadequate FWLCS response was a non-conforming condition with the above description in the UFSAR.

The flow of feedwater from the two turbine driven reactor feedwater pumps to the reactor vessel is controlled by the FWLCS. A master level controller normally provides overall control of the individual reactor feedwater pumps' flow controllers and is normally set up by plant operators to automatically maintain the desired reactor vessel water level. The master controller maintains reactor vessel water level by varying feedwater system flow based upon actual feedwater flow, steam flow, and reactor vessel water level. The master controller is designed not only to respond to actual differences in the above three parameters, but also to their rate of change in order to anticipate and limit the change in reactor vessel water level and to return it to the programmed level. The system is designed to be an "inventory dominant system," meaning that changes in feedwater flow and steam flow predominately affect the response of the system. Level and level error provide inputs to the secondary control loop. The system should provide an integral response to changes in feedwater flow, steam flow, and reactor vessel water level, such that the master level controller output will increase with the duration and rate of change in these parameters.

Review of plant data for the event revealed that as reactor vessel water level rose in response to the recirculation pump trip, FWLCS output to the feedwater system decreased. However, the output of the master level controller only decreased in response to lowering steam flow. There was not an appropriate response to the increasing difference between the actual reactor vessel water level and the desired reactor vessel water level. As the transient proceeded, the reactor vessel water level remained above the desired level so that the rate of change for the master level controller output should have increased. Additionally, since the difference in the actual level and the desired level was increasing, the master level controller should have been reducing its output even more for a maximum reduction in feedwater flow.

On February 10, 2008, Unit 1 automatically scrammed following an unexpected trip of the 'B' reactor recirculation pump. The loss of the reactor recirculation pump caused reactor vessel water level to increase, resulting in a high reactor vessel water level (Level 8) scram. Review of the plant process computer traces from the February 2008 reactor scram revealed the same FWLCS response that occurred on October 15, 2009. The inspectors reviewed the cause for the February 2008 scram in NRC Inspection Report 05000461/2008004 and concluded that the licensee had failed to perform adequate post-maintenance testing following replacement of the FWLCS dynamic compensator card during the Cycle 10 refueling outage in February 2006. The licensee restarted Unit 1 following the February 2008 reactor scram without identifying and correcting the cause of the scram. In hindsight, the problem with the FWLCS response was not well understood by the licensee when the unit was restarted.

During review of the February 2008 event, the inspectors discussed a concern with licensee management regarding an apparent lack of technical rigor in evaluating the plant start up after the reactor scram and continued operation of the unit with this non-conforming condition because very little was documented regarding its impact on the plant and on the operators' ability to respond to reactor vessel water level transients. There was no formal evaluation of the non-conforming condition performed and there was no justification documented for starting up the unit and operating for a 2-year cycle with the non-conforming condition. The inspectors discussed this issue with the plant manager and AR 00807670 was written in August 2008 for the engineering staff to perform an evaluation. The licensee subsequently performed the evaluation using its Operations Technical Decision Making (OTDM) Process in October 2008. The decision made was to continue plant operation with a deficient FWLCS response and to replace specific circuit components during a forced outage or next refueling outage, and to perform dynamic tuning of the control system online after replacement of the circuit components. The licensee believed that this was a good decision because:

1. Completing the system dynamic tuning would ensure that the FWLCS responds adequately and is in compliance with the UFSAR during transient conditions.

2. The decision would eliminate an additional plant shutdown and the associated risks.

3. The risk of a reactor scram from the current condition of the FWLCS was suitably low.

4. The current condition did not impact normal plant operation and the plant had remained stable during scheduled down-powers and rod sequence exchanges.

During review of the February 2008 event, the inspectors concluded that, although the non-conforming condition had not been corrected, the issue was not a significant safety concern because it was bounded by the UFSAR, Chapter 15 safety analyses that include a reactor scram from the trip of both reactor recirculation pumps.

On May 30, 2009, the licensee reduced power to about 28 percent to implement temporary alterations to out-of-service Turbine Building Heater Bay room fan coolers and to attempt repair of a packing leak from a feedwater system valve in the Heater Bay. The licensee had intended to perform data collection needed by the vendor to support tuning of the FWLCS; however, an unexpected reactor water level transient during power ascension occurred when transferring one of the turbine driven reactor feedwater pumps from manual to automatic control on the master level controller and the licensee aborted the data collection activity. After the May 2009 transient, the plan for data collection and tuning was not a high priority for the licensee. On September 10, 2009, the licensee completed a revision to its OTDM plan to execute data collection. The plan called for an iterative approach to fine tuning the FWLCS that would start with online tuning of the individual feedwater flow and master level controllers, an adjustment of the flow controllers and the master level controller during an outage and comparison of the tuning after startup from the outage.

On September 29, 2009, the licensee shut down Unit 1 to repair a steam leak on the RCIC system inboard steam supply isolation valve. The unit was restarted and synchronized to the grid on October 4th. During the forced outage, the licensee replaced several FWLCS circuit components and performed post-maintenance testing.

However, data collection needed to support tuning of the FWLCS was not performed during power ascension as planned. This forced outage occurred just two weeks before the reactor scram on October 15th. Because of the perceived low risk for another reactor scram before the January 2010 refueling outage, the licensee did not correct the non-conforming condition before starting up from the forced outage.

The inspectors thoroughly examined the licensee's root cause evaluation for the reactor recirculation pump motor failure and concluded that the licensee had not neglected any likely factors. The licensee shipped the motor to a vendor for failure analysis. The vendor performed testing and examination of the motor and concluded that the apparent cause of the failure was a random insulation breakdown of the original motor windings that resulted in a turn-to-turn failure. The failure was limited to a single coil end turn knuckle of the winding and was located in a position that was not susceptible to air flow or foreign material. The motor was purchased and maintained as a station spare since initial plant construction. It was refurbished in December 2003 and installed in the plant in February 2004. The motor windings were the original windings. The inspectors did not identify a performance deficiency involving the pump motor failure.

The inspectors reviewed the licensee's apparent cause evaluation that was performed to investigate the organizational decision making involved with the licensee's treatment of the non-conforming FWLCS. There was one apparent cause and one contributing cause identified by the licensee:

1. There was a long lead time to replace the circuit components and tune the FWLCS, which dated back to inadequate post-maintenance testing of the system in February 2006. The licensee originally believed that since circuit cards were pre-calibrated to match the previously installed cards, nothing was changed when the cards were replaced and therefore dynamic tuning was not necessary. Furthermore, it was not recognized that engineering technical resource limitations hindered the resolution of the issue and prevented the licensee from obtaining a contractor to assist in resolving the issue. A plan to execute data collection and system tuning was not a high priority for the licensee and engineering development of some technical evaluations was incomplete or ineffective. As a result, the non-conforming condition was left uncorrected. (apparent cause)

2. Because of the perceived low risk for FWLCS response causing another reactor scram, fine tuning was not an immediate priority for the licensee. (contributing cause)

The licensee identified two corrective actions in the apparent cause evaluation:

1. Provide training to engineering personnel on decision making. This will include training on the decision making process, critical thinking/probing, and challenging assumptions.

2. Provide a case study as part of the annual Outage Control Center and Scope Committee training.

The licensee had not yet completed the above corrective actions at the conclusion of this inspection.

On November 26, 2009, the licensee completed an OTDM plan to determine when to perform fine tuning of the FWLCS. The decision made was to perform fine tuning of the system up to the intermediate limit of the flow/level controllers during coast down just prior to the January 2010 refueling outage, with controller setpoint range allowance changes during the refueling outage and dynamic tuning completed during power ascension after the refueling outage. The FWLCS response has been corrected. Dynamic tuning of the system was completed in January 2010 and system response was verified by the licensee upon start up from the January-February 2010 refueling outage. Refer to Section 4OA3.1 of this inspection report for a review and closure of the Licensee Event Report (LER) associated with the reactor scram.

The licensee's Maintenance Rule Expert Panel reviewed the performance of the FWLCS as a result of the February 2008 and October 2009 reactor scrams and concluded that the plant level reliability performance criteria were exceeded when the second reactor scram functional failure occurred against the limit of one failure in a rolling 24-month period. The second failure was determined to be a repetitive maintenance preventable functional failure, for which the licensee established goals and implemented a monitoring plan in accordance with Paragraph (a)(1) of the Maintenance Rule. The inspectors reviewed the licensee's monitoring plan and did not identify any issues of concern.

<u>Analysis</u>

The inspectors determined that the failure to correct the non-conforming condition with the FWLCS response that resulted in a second reactor scram for the same cause 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 determined that there were no examples related to this issue. Consistent with the guidance in IMC 0612, Appendix B, "Issue Screening," the inspectors determined that the finding was of more than minor significance because this issue was associated with the Equipment Performance attribute of the Initiating Events Cornerstone and adversely affected the cornerstone objective of limiting the likelihood of events that upset plant stability and challenge critical safety functions during power operations. Specifically, inadequate FWLCS response resulted in a reactor scram following the unexpected loss of a reactor recirculation pump. 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, MS, and BI Cornerstones," the inspectors determined that that this finding was a licensee performance deficiency of very low safety significance (Green) because the finding: (1) did not contribute to the likelihood of a LOCA initiator, (2) did not contribute to <u>both</u> the likelihood of a reactor trip AND the likelihood that mitigation equipment or functions would not be available, and (3) did not increase the likelihood of a fire or internal/external flooding event.

Cross-Cutting Aspects

The inspectors did not identify a cross-cutting aspect related to this finding.

Enforcement

No violation of regulatory requirements was identified. This issue is considered to be a finding **(FIN 05000461/2010002-05, Failure to Correct Inadequate FWLCS Response Resulted in High Reactor Vessel Water Level Scram)**. The licensee entered this finding into its corrective action program as AR 00983144.

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

a. Inspection Scope

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

- Emergent maintenance on January 13th to restore the Emergency Reserve Auxiliary Transformer Static VAR [Volt-Ampere Reactive] Compensator to service following a cooling pump failure;
- Planned maintenance on March 9th on the Motor Driven Reactor Feedwater Pump;
- Planned maintenance associated with the C1R12 Refueling Outage (shutdown safety risk review for the outage schedule); and
- Planned maintenance to replace Control Rod Drive Mechanisms during the refueling outage.

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.

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 four maintenance risk assessment inspection samples as defined in IP 71111.13.

b. Findings

(1) Operations with the Potential to Drain the Reactor Vessel

Introduction

The inspectors reviewed evaluation CL-2010-E-001, "CPS Operations with the Potential to Drain the Reactor Vessel (OPDRV)," Revision 0, for compliance with 10 CFR 50.59, "Changes, Tests, and Experiments." The inspectors have questions remaining whether the implementation of the new procedure would create the possibility for an accident of a different type than any previously evaluated in the UFSAR. This issue is considered to be an Unresolved Item pending additional review by the inspectors.

Discussion

On January 11, 2010, the licensee approved CPS 3711.01, which established the definition of an OPDRV for use in determining the applicability of several TS requirements while in Modes 4 and 5. Prior to the procedure approval, an evaluation, CL-2010-E-001, was performed by the licensee to address the criteria of 10 CFR 50.59(c)(2). In its evaluation, the licensee concluded that none of the eight criteria were applicable under this requirement to obtain a license amendment pursuant to 10 CFR 50.90 prior to implementing the new procedure. Specifically, Criterion (v) which requires a license amendment prior to implementing a change that creates a possibility for an accident of a different type than any previously evaluated in the UFSAR was determined to not apply.

Upon review of the implementation of CPS 3711.01, the inspectors identified a number of credible accident scenarios that were introduced by this procedure change and not previously evaluated in the UFSAR. The Clinton UFSAR analyzed a LOOP for only at-power conditions. This LOOP could also occur in Modes 4 and 5 at similar frequency and could have significance due to the implementation of the new procedure. Evolutions with non-isolable RCS leakage below 5 gpm, and up to 420 gpm with a single method of isolation were made procedurally acceptable and not to be considered an OPDRV by the licensee. Evolutions with RCS leakage up to these values could occur simultaneously with the inoperability of all ECCS and still remain in compliance with TS 3.5.2, which addresses the ECCS requirements for Modes 4 and 5. In addition, coincident with this scenario, neither primary nor secondary containment was required to be set according to TS 3.6.1.1 and 3.6.4.1, respectively. If a LOOP were to occur in this condition during refueling, it could possibly be of significance. No evaluation was performed by the licensee to determine how much time might be available for operators to respond prior to uncovering irradiated fuel in the upper containment pool for such a scenario. Rough estimations performed by the inspectors found the available time for operator response to be no more than six hours. In EC 376912, the licensee evaluated the time it would take to drain down to the reactor vessel flange with the new RCS leakage limits defined in CPS 3711.01 and found there to be greater than 17 hours for operators to respond. This evaluation did not consider the potential for uncovering irradiated fuel in the upper containment pool. The licensee's evaluation accepted a "non-controversial basis in the industry" in defining an OPDRV. That definition specified that an open penetration that has the potential to uncover irradiated fuel is an OPDRV. Irradiated fuel would necessarily include any fuel that could be uncovered in the upper containment pool.

Other credible scenarios introduced by the procedure change include the possible failure of administrative controls to preclude inadvertent withdrawal of a control rod that is being relied upon to isolate leakage for a removed control rod drive mechanism; or a fuel handling accident, which could result in a load movement disturbing the proper seating of such a control rod. The inspectors noted previous examples of failures of administrative controls similar to those relied upon in CPS 3711.01 to eliminate the potential for an accident. Notably during the recent refueling outage, on January 18, 2010, a location error occurred when the licensee's refueling team discovered they had removed the incorrect control rod double blade guide. This incident occurred despite several procedural and human performance error prevention tools that were utilized. In addition, during the February 2008 refueling outage only a good questioning attitude by a contractor prevented removing safety relief valves prior to installing main steam line plugs and possibly resulting in the loss of RCS inventory. The inspectors requested assistance from regional specialists to review the licensee's 10 CFR 50.59 evaluation and procedure change. This issue is considered to be an Unresolved Item **(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)** pending additional review. The licensee wrote AR 01051306 to address the inspectors' questions through its corrective action program.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- AR 00976295, "ECCS Floor Drain Piping Connected to the Radwaste Pipe Tunnel;"
- AR 01017629, "Excess Flow Check Valve (1SM008) Slow to Reopen;"
- AR 01032794, "Diesel Generator Fuel Oil Consumption Rate;"
- AR 01008721, "9069.01 Flow Data Less Than Acceptable;" and
- AR 01042588, "1VP04CB: Division 2 VP/WO [Drywell Cooling/Chilled Water] Isolation – Shunt Trip."

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 five operability evaluation inspection samples as defined in IP 71111.15.

b. Findings

No findings of significance were identified.

1R18 Plant Modifications (71111.18)

.1 Temporary Modifications

a. Inspection Scope

The inspectors reviewed the following temporary plant modification:

- Engineering Change (EC) 377772, "Jumper Out High Compressor Oil Temperature Trip on 1VP04CB."

The inspectors reviewed the temporary modification and the associated 10 CFR 50.59 screening/evaluation against applicable system design basis documents, including the UFSAR and the TS to verify whether applicable design basis requirements were satisfied. The inspectors reviewed the operator logs and interviewed engineering and operations department personnel to understand the impact that implementation of the temporary modification had on operability and availability of the affected plant equipment.

The inspectors also reviewed a sample of action requests pertaining to temporary modifications to verify that problems were entered into the licensee's corrective action program with the appropriate significance characterization and that corrective actions were appropriate.

This inspection constituted one temporary modification inspection sample as defined in IP 71111.18.

b. Findings

No findings of significance were identified.

.2 Permanent Modifications

a. Inspection Scope

The inspectors reviewed the engineering analyses, modification documents, and design change information associated with the following permanent plant modifications:

- EC 369979, "Line Stop on Line 1WS33BA-20" Cross Tie Between Plant Service Water (WS) and Division 1 Shutdown Service Water (SX) to Repair Motor Operated Valve (MOV) 1SX014A;" and
- EC 371560, "Installation of High Point Vent on Line 1HP02A-14."

During this inspection, the inspectors evaluated the implementation of the design modifications and verified, as appropriate, that:

- the compatibility, functional properties, environmental qualification, seismic qualification, and classification of materials and replacement components were acceptable;
- the structural integrity of the SSCs would be acceptable for accident/event conditions;
- the implementation of the modification did not impair key safety functions;
- no unintended system interactions occurred;
- the affected significant plant procedures, such as normal, abnormal, and emergency operating procedures, testing and surveillance procedures, and training were identified and necessary changes were completed;
- the design and licensing documents were either updated or were in the process of being updated to reflect the modification;
- the changes to the facility and procedures, as described in the UFSAR, were appropriately reviewed and documented in accordance with 10 CFR 50.59;
- the system performance characteristics, including energy needs affected by the modification continued to meet the design basis;
- the modification test acceptance criteria were met; and
- the modification design assumptions were appropriate.

Completed activities associated with the implementation of the modification, including testing, were also inspected, and the inspectors discussed the modification with the responsible engineering and operations staff.

Additional activities were performed during the evaluation of EC 371560 that were associated with TI 2515/177, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.” These activities are described in Section 1R18.3 of this inspection report.

This inspection constituted two permanent modification inspection samples as defined in IP 71111.18.

b. Findings

No findings of significance were identified.

.3 Permanent Plant Modifications associated with TI 2515/177, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems.”

a. Inspection Scope

The following engineering design package associated with the scope of GL 2008-01, “Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems,” was reviewed and selected aspects were discussed with engineering personnel:

- EC 371560, "Installation of High Point Vent on Line 1HP02A-14."

The inspectors verified that the licensing basis verification documents have either been updated, or are in the process of being updated, to reflect the modifications associated with the licensee's resolution of GL 2008-01 (TI 2515/177, Section 04.01). The verified documents included the TS, TS Bases, UFSAR, and licensee-controlled documents and bases, such as the Technical Requirements Manual.

In addition, the inspectors verified that the drawings were up-to-date with respect to recent hardware changes and that any discrepancies between as-built configurations and the drawings were documented and entered into the corrective action program for resolution (TI 2515/177, Section 04.02.a.6).

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

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

b. Findings

No findings of significance were identified.

1R19 Post-Maintenance Testing (71111.19)

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:

- Work Order (WO) 0000266417, "1B21F022B: Post-Maintenance Test for Overhaul Per CPS 8216.11;"
- WO 0000266415, "1B21F022B: Local Leak Rate Test Retest Per CPS 9861.01D002;"
- WO 0000266504, "1B21F022C: Post-Maintenance Test for Overhaul Per CPS 8216.11;"
- WO 0091849420, "1SX01PB: Perform SX Pump Operability Test" and WO 0091849410, "1SX01PB: PMT-Collect Vibration/Thermography;"
- WO 0001130746, "Perform Breaker Installation Support and Functional Testing of New Breaker Installed at 1RR02ED;"
- WO 1303779-04, "1SM008: Leak Rate Test Retest;"
- WO 00790605-03, "1E21F006: Leak Rate Test Retest;"
- WO 00750261-04, "1E12F025A: Local Leak Rate Test Retest;" and
- WO 01308289, "9067.01 OP SGTS Train Flow/Heater Operability Run (Train A)."

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 nine post-maintenance testing inspection samples as defined in IP 71111.19.

b. Findings

No findings of significance were identified.

1R20 Outage Activities (71111.20)

.1 Unit 1 Refueling Outage (C1R12)

a. Inspection Scope

The inspectors evaluated the licensee's conduct of C1R12 refueling outage activities to assess the licensee's control of plant configuration and management of shutdown risk. The inspectors reviewed configuration management to verify that the licensee maintained defense-in-depth commensurate with the shutdown risk plan; reviewed major outage work activities to ensure that correct system lineups were maintained for key mitigating systems; and observed refueling activities to verify that fuel handling operations were performed in accordance with the TSs and approved procedures. Other major outage activities evaluated included the licensee's control of the following:

- containment penetrations in accordance with the TSs;
- SSCs that could cause unexpected reactivity changes;
- flow paths, configurations, and alternate means for RCS inventory addition;
- SSCs (e.g., control rod drive mechanism replacements) that could cause a loss of inventory;
- RCS level instrumentation;
- radiological work practices;
- spent fuel pool cooling during and after core offload;
- switchyard activities and the configuration of electrical power systems in accordance with the TSs and shutdown risk plan; and
- SSCs required for decay heat removal and for establishing alternate means for decay heat removal, including instrumentation.

The inspectors observed portions of the plant cooldown, including the transition to shutdown cooling to verify that the licensee controlled the plant cooldown in accordance with the TSs. The inspectors also observed portions of the restart activities including plant heat up to verify that TS requirements and administrative procedure requirements were met prior to changing operational modes or plant configurations. Major restart inspection activities performed included:

- verification that RCS boundary leakage requirements were met prior to entry into Mode 3 and subsequent operational mode changes;
- ECCS filling and venting to ensure no large air voids remained that could affect ECCS pump performance during LOCA conditions;
- verification that primary and secondary containment integrity was established prior to entry into Mode 3; and
- inspection of the Containment Building, including the drywell, to assess material condition and search for loose debris, which if present could block floor drains in the drywell or be transported to the containment suppression pool.

The inspectors interviewed operations, engineering, work control, radiological protection, and maintenance department personnel and reviewed selected procedures and documents.

In addition, the inspectors reviewed a sample of issues that the licensee entered into the corrective action program related to outage activities to verify that identified problems were being entered with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for refueling outage issues documented in selected action requests.

This inspection constituted one refueling outage inspection sample as defined in IP 71111.20.

b. Findings

No findings of significance were identified.

.2 Unit 1 Forced Outage (C1F53)

a. Inspection Scope

The inspectors evaluated outage activities during Unit 1 forced outage C1F53, which began on February 5, 2010. During the performance of turbine on-line testing with the unit 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. The reactor remained on line while troubleshooting the cause for the turbine trip. The licensee synchronized the unit to the grid on February 8th.

The inspectors reviewed plant equipment configuration, risk management, startup activities, and identification and resolution of problems associated with the outage.

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

b. Findings

No findings of significance were identified with the conduct of forced outage activities.

During the refueling outage, the turbine mechanical trip device trip finger plunger gap was mis-adjusted by the turbine vendor while performing an inspection. This mis-adjusted gap caused the turbine trip logic to activate prematurely during start-up testing, causing the forced outage. This issue is considered to be an Unresolved Item **(URI 05000461/20100002-07, Main Turbine Trip During On-line Testing)** pending the inspectors' review of the licensee's cause evaluation for the event.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors reviewed the test results for the following surveillance testing activities 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 9864.01D004, "Low Pressure Excess Flow Check Valve Test Data Sheet for 1SM008;" (IST)
- CPS 9069.01, "Shutdown Service Water Operability Test;" (IST)
- CPS 9861.04, "MSIV [Main Steam Isolation Valve] Local Leak Rate Test [LLRT] (MC-5,6,7,8);"(LLRT)
- CPS 9861.02D014, "LLRT Data Sheet for 1MC036 – LPCS Injection;" (LLRT)
- CPS 9861.05D002, "LPCS Water Test (CLOC);" (LLRT)
- CPS 9059.01, "Reactor Coolant System Leakage Test;" (RCS Leakrate)
- CPS 9843.01V019, "RHR 'B' Keep Fill for 1E12F495A;"
- CPS 9053.05, "Containment Spray System Functional Test" (for Train 'A');
- CPS 9080.23, "Diesel Generator 1C – ECCS Integrated;"
- CPS 9080.22, "Diesel Generator 1B – ECCS Integrated;" and
- MA-AA-723-300, "Diagnostic Testing of Motor Operated Valves" (As-Found VOTES Test for 1E12F042A).

The inspectors observed selected portions of the test activities 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 two ISTs, three containment isolation valve LLRTs, one RCS leakrate detection test, and five routine surveillance tests for a total of eleven surveillance testing inspection samples as defined in IP 71111.22.

b. Findings

(1) Inadequate Test Criteria in Standby Gas Treatment (SGT) System Flow/Heater Operability Surveillance Test

Introduction

The inspectors identified a finding of very low safety significance (Green) with an associated Non-Cited Violation of 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings." The licensee failed to include appropriate quantitative or qualitative acceptance criteria in its surveillance test procedure for fulfilling the monthly Surveillance Requirement (SR) to demonstrate operability of the SGT system.

Discussion

The inspectors reviewed the licensee's performance of surveillance testing that was accomplished in accordance with procedure CPS 9067.01, "Standby Gas Treatment System Flow/Heater Operability," Revision 31a. This surveillance test procedure was performed to satisfy TS SR 3.6.4.3.1, which required that each SGT subsystem (or train) be operated for ≥ 10 continuous hours with the heaters operating once every 31 days. As described in the UFSAR, the safety function of the SGT is to minimize the offsite release of radioactive materials that leak from the primary containment into the secondary containment following a design basis accident to limit the offsite and Control Room dose to the guidelines of 10 CFR 50.67.

According to the Bases for TS SR 3.6.4.3.1: "Operating each SGT subsystem from the main control room for ≥ 10 hours ensures that both subsystems are operable and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and high efficiency particulate air (HEPA) filters." The Bases also states that: "The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system."

During review of CPS 9067.01, the inspectors noted that the procedure did not have specific steps to ensure that flow blockage did not exist by verifying that each SGT subsystem provided sufficient air flow. The acceptance criteria specified in Step 9.1.1 of the procedure required only that the "SGT train operates for ≥ 10 hours with flow through the train and the heater is operable." Although SGT subsystem inlet flow was recorded, there were no acceptance criteria in the procedure to evaluate whether each subsystem was capable of providing the minimum required air flow to meet its safety function. According to the UFSAR, the SGT system was designed with a flow control valve that maintains flow at 4000 cubic feet per minute (± 10 percent); however, there was no comparison of the recorded flow rates with the design flow rate to ensure that the fan and/or the flow control valve were operating properly or that there was no flow blockage. Although pre-filter and HEPA filter differential pressures were recorded, the acceptance criteria provided in Step 9.2.1 of CPS 9067.01 only established criteria for dirty filter replacements. The criteria were not used to evaluate whether each subsystem was capable of providing the minimum required air flow to meet its safety function. Although SGT subsystem inlet and outlet temperatures were recorded three times during the

10-hour run, there were no acceptance criteria in the procedure to evaluate whether the heater was capable of providing sufficient heat to eliminate moisture on the adsorbers and HEPA filters. There were also no specific steps in CPS 9067.01 to measure and evaluate fan and motor vibration levels or to locally assess the running subsystem for abnormalities. Local inspection of the subsystem during operation (e.g., checking rotating equipment for abnormal temperatures, odors, noise and/or vibration) would ensure that blockage, fan or motor failure, or excessive vibration could be detected for corrective action.

Analysis

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 surveillance testing requirement was a performance deficiency warranting a significance evaluation. 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, "Power Reactor Inspection Reports, Appendix B, "Issue Screening," the inspectors determined that the finding was associated with the Procedure Quality Cornerstone attribute for the Control Room and Auxiliary Building and adversely affected the Barrier Integrity Cornerstone objective to provide reasonable assurance that physical design barriers protect the public from radionuclide releases caused by accidents or events. Specifically, by not providing appropriate acceptance criteria by which the operability of the SGT system trains could be assessed, the ability of the SGT system to collect and treat the design leakage of radionuclides from the primary containment to the secondary containment during an accident could not be assured. 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, MS, and BI Cornerstones," the inspectors determined that that this finding was a licensee performance deficiency of very low safety significance (Green) because the finding involved only a degradation of the radiological barrier function provided by the SGT system.

Cross-Cutting Aspects

The inspectors concluded that because the performance issue was associated with procedure changes implemented during carryover from original TS into improved standard TS, it did not necessarily reflect current licensee performance and no cross-cutting aspect was identified.

Enforcement

Title 10 CFR 50, Appendix B, Criteria V, "Instructions, Procedures, and Drawings" requires 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, the licensee failed to provide appropriate quantitative or qualitative acceptance criteria in surveillance test procedure, CPS 9067.01, "Standby Gas Treatment System Flow/Heater Operability," Revision 31a, to demonstrate the operability of the SGT system as described in the TS Bases, an activity affecting quality. Specifically, the procedure did not have specific steps to verify that each SGT subsystem provided sufficient air flow and to verify whether the heater was capable of providing sufficient heat to eliminate moisture on the adsorbers and HEPA filters. In addition, the procedure did not have steps to inspect or evaluate fan and motor vibration levels or to locally assess the running subsystem for abnormalities. Because of the very low safety significance, this violation is being treated as a Non-Cited Violation consistent with Section VI.A of the NRC Enforcement Policy **(NCV 05000461/2010002-08, Inadequate Test Criteria in Standby Gas Treatment System Flow/Heater Operability Surveillance Test)**. The licensee entered this violation into its corrective action program as AR 01023451.

2. RADIATION SAFETY

Cornerstone: Occupational Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

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

- ISI Activities Inside the Bio-shield;
- Drywell – Scaffolding;
- Drywell Shielding;
- Drywell – Safety Relief Valve (SRV) Work;
- RT Pipe Replacement; and
- Refuel Floor Work.

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

- identification of hot particles;
- the presence of alpha emitters;
- the potential for airborne radioactive materials, including the potential presence of transuranics and/or other hard-to-detect radioactive materials);

- the hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee established a means to inform workers of changes that could significantly impact their occupational dose; and
- severe radiation field dose gradients that can result in non-uniform exposures of the body.

This inspection constitutes a partial sample as defined in IP 71124.01.

b. Findings

No findings of significance were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

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

- RWP 10010079; C1R12 Drywell – ISI Inside Bio-shield;
- RWP 10010087; C1R12 Drywell – Scaffolding;
- RWP 10010088; C1R12 Drywell Shielding;
- RWP 10010098; C1R12 Drywell – SRV Work;
- RWP 10010101; RT Pipe Replacement; and
- RWP 10010150; C1R11 [sic] Refuel Floor Work.

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

b. Findings

No findings of significance were identified.

.3 Radiological Hazards Control and Work Coverage (02.05)

a. Inspection Scope

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

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

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

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

The inspectors reviewed the following RWPs for work within airborne radioactivity areas with the potential for individual worker internal exposures.

- RWP 10010079; C1R12 Drywell – ISI Inside Bio-shield;
- RWP 10010098; C1R12 Drywell – SRV Work;
- RWP 10010101; RT Pipe Replacement; and
- RWP 10010150; C1R11 [sic] Refuel Floor Work.

For these RWPs, the inspectors evaluated airborne radioactive controls and monitoring, including potentials for significant airborne levels (e.g., grinding, grit blasting, system breaches, entry into tanks, cubicles, reactor cavities). The inspectors assessed barrier (e.g., tent or glove box) integrity and temporary HEPA ventilation system operation for selected airborne radioactive material areas.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors discussed with the Radiation Protection Manager (RPM) the controls and procedures for high-risk HRAs and Very High Radiation Areas (VHRAs). The inspectors assessed whether any changes to licensee procedures substantially reduced the effectiveness and level of worker protection.

The inspectors reviewed special areas that have the potential to become VHRAs during certain plant operations (e.g., BWR drywell fuel transfer slot area; spent fuel pool; cavity; or pit diving). The inspectors discussed these areas with first-line health physics (HP) supervisors (or equivalent positions having backshift HP oversight authority) to assess the communication beforehand with the HP group to determine if it would allow for corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

b. Findings

No findings of significance were identified.

.5 Radiation Worker Performance (02.07)

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

.6 Radiation Protection Technician Proficiency (02.08)

a. Inspection Scope

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

b. Findings

No findings of significance were identified.

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

This inspection constitutes a partial sample as defined in IP 71124.02.

.1 Radiological Work Planning (02.02)

a. Inspection Scope

The inspectors assessed whether the licensee's planning identified appropriate dose mitigation features; considered alternate mitigation features; and defined reasonable dose goals. The inspectors evaluated the licensee's As-Low-As-Is-Reasonably-Achievable (ALARA) assessment to determine if it had taken into account decreased worker efficiency from use of respiratory protective devices and or heat stress mitigation equipment (e.g., ice vests). The inspectors determined whether the licensee's work planning considered the use of remote technologies (such as teledosimetry, remote visual monitoring, and robotics) as a means to reduce dose, the use of dose reduction insights from industry operating experience, and plant-specific lessons learned. The inspectors assessed the integration of ALARA requirements into work procedure and RWP documents.

b. Findings

No findings of significance were identified.

.2 <u>Verification of Dose Estimates and Exposure Tracking Systems</u>

a. <u>Inspection Scope</u>

The inspectors assessed whether the licensee had established measures to track, trend, and if necessary, to reduce occupational doses for ongoing work activities. The inspectors assessed whether trigger points or criteria were established to prompt additional reviews and/or additional ALARA planning and controls.

b. <u>Findings</u>

No findings of significance were identified.

.3 <u>Radiation Worker Performance</u> (02.05)

a. <u>Inspection Scope</u>

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

b. <u>Findings</u>

The inspectors identified an Unresolved Item concerning events that occurred on January 17, 2010, when a radiation protection technician (RPT) observed a radiation worker with an arm inside a highly radioactive contaminated system without the proper equipment to control the spread of radioactive contamination. The radiation worker remained inside the drywell and performed additional work, after the RPT made this observation. Subsequently, the radiation worker was allowed to leave the Radiologically Controlled Area (RCA) without successfully passing a contamination monitor and without additional radiological controls. Radioactive contamination was later identified when the radiation worker caused contamination monitor alarms. At the time of the inspection, the radiation worker's access to the RCA was restricted and the licensee was still in the process of compiling all of the facts surrounding the event(s). Additionally, an evaluation of the radiological hazards of the work performed was still being developed. As a result, any radiological consequence for the worker was unknown. Similarly, the inspectors could not evaluate the consequence of any apparent improper radiological controls. Therefore, this issue remains under review by the NRC and is categorized as an Unresolved Item **(URI 05000461/2010002-09; Individual Contaminated in RT Hold Pump Room)**.

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 Fourth Quarter 2009 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 Unplanned Scrams per 7000 Critical Hours

a. Inspection Scope

The inspectors verified the Unplanned Scrams per 7000 Critical Hours Performance Indicator for Unit 1. The inspectors reviewed each LER from January 1, 2009, through December 31, 2009, determined the number of scrams that occurred, and verified the licensee's calculation of critical hours. The inspectors also reviewed the licensee's corrective action program database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

This inspection constituted one performance indicator verification inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

.3 Unplanned Scrams with Complications

a. Inspection Scope

The inspectors verified the Unplanned Scrams with Complications Performance Indicator for Unit 1. The inspectors reviewed each LER from January 1, 2009, through December 31, 2009, determined the number of scrams that occurred, and evaluated each of the scrams against the performance indicator definition. The inspectors also reviewed the licensee's corrective action program database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

This inspection constituted one performance indicator verification inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

.4 Unplanned Transients per 7000 Critical Hours

a. Inspection Scope

The inspectors verified the Unplanned Transients per 7000 Critical Hours Performance Indicator for Unit 1. The inspectors reviewed power history data from January 1, 2009, through December 31, 2009, determined the number of power changes greater than 20 percent full power that occurred, evaluated each of the power changes against the performance indicator definition, and verified the licensee's calculation of critical hours. The inspectors also reviewed the licensee's corrective action program database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator.

This inspection constituted one performance indicator verification inspection sample as defined in IP 71151.

b. Findings

No findings of significance were identified.

.5 Safety System Functional Failures

a. Inspection Scope

The inspectors verified the Safety System Functional Failures Performance Indicator for Unit 1. The inspectors reviewed each LER from January 1, 2009, through December 31, 2009, determined the number of safety system functional failures that occurred, evaluated each LER against the performance indicator definition, and verified the number of safety system functional failures reported. The inspectors also reviewed the licensee's corrective action program database to determine if any problems had been identified with the performance indicator data collected or transmitted for this indicator and none were identified.

This inspection constituted one performance indicator verification 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 containment suppression pool. The inspectors opened URI 05000461/2009004-01 to review the licensee’s evaluation of the condition. This issued 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. Unresolved Item 05000461/2009004-01 remains open pending the resolution of questions posed by the inspectors regarding the licensee's evaluation of the extent of condition and reporting requirements.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review of Identification and Resolution of Problems

a. Inspection Scope

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 Annual In-Depth Review Samples

a. Inspection Scope

The inspectors selected the following issue for in-depth review:

- EC 0000376171, "CPS Fire Protection Program: Evaluation of TCFZs and Determination of Negligible Quantities of Combustible Materials and Their Impact on TCFZs," Revision 0

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

b. Findings and Observations

(1) Non-Conservative Fire Protection Program Change

The inspectors reviewed EC 0000376171, which established criteria to address how the licensee's engineering and fire protection staff would evaluate the presence of transient combustible materials in transient combustible free zones (TCFZs). The licensee prepared this engineering evaluation in July 2009 to define and justify the use of "negligible quantities of combustibles" in TCFZs. The intent of EC 0000376171 was to define an amount of transient combustible materials that could be considered "negligible" for the purposes of meeting the criteria of no intervening combustibles or an area devoid of combustibles. The engineering evaluation stipulated that transient combustible materials that are to be intentionally left in a TCFZ shall be required to have a transient combustible permit (TCP) approved by the Site Fire Marshall and if the specified criteria cannot be met, then a Plant Barrier Impairment and additional compensatory measures shall be applied. The licensee also revised the Clinton Power Station site specific information within the Exelon corporate procedure, OP-AA-201-009, “Control of Transient Combustible Material,” Revision 10, and Clinton Power Station procedure OP-CL-201-009, “Control of Transient Combustible Material,” Revision 0, to incorporate the criteria established by EC 0000376171.

The criteria within EC 0000376171, which delineated what was considered a negligible quantity of material, included:

- The amount of identified transient combustible materials did not change the overall fuel load classification for the area.

- The amount of identified transient combustible materials did not impact fire barrier justifications.

- The amount of identified transient combustible materials was limited to a calculated 1000 British Thermal Units (Btu) per square foot (ft^2) or 700 Btu/ft^2, depending on the TCFZ, over the area of the TCFZ.

The inspectors found that the criterion limiting identified transient combustibles to a specified value taken over the entire area of a TCFZ could result in a non-conservative value being determined for what was intended to be a negligible fuel load. For example, a TCFZ was located above the Chemistry Laboratory Offices on the 751’0” elevation of the Control Building within Fire Zone CB-1e. The TCFZ provided divisional separation between two redundant trains of cables required for safe shutdown. The inspectors noted that EC 0000376171 specified that 700 Btu/ft^2 was considered negligible for this TCFZ. Specifically, the licensee determined that 2,041,900 Btu of energy for the 2,917 ft^2 area of the TCFZ would be a negligible quantity. The inspectors determined that 2,041,900 Btu of energy was equivalent to an approximate 1.2 megawatt heat release rate over a 30 minute period. Compounding the significance, the inspectors noted that much of this TCFZ had a relatively low ceiling (approximately 10 feet) and the TCFZ contained exposed cables. Using the spreadsheets in NUREG-1805, “Fire Dynamics Tools (FDTs) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program,” the inspectors determined that a heat release rate from a 1.2 megawatt fire would result in plume and ceiling jet temperatures sufficient to damage and ignite unprotected cables within the TCFZ. The ignition of in-situ combustibles, such as cables, would further contribute to the overall heat release rate and challenge the effectiveness of divisional separation within the area; and, as a result, would challenge the ability to achieve and maintain safe shutdown. The inspectors noted that a wet-pipe sprinkler system did provide suppression coverage for the TCFZ on one side; however, the suppression coverage provided one element of defense-in-depth in addition to the intended divisional separation and was not to be relied upon exclusively.

The inspectors noted that the general guidelines established in Section 4.4 of OP-CL-201-009 required that any transient combustible materials within a TCFZ be constantly attended or limited. However, Attachment 4 to OP-CL-201-009 contained provisions for materials to be authorized under a TCP to be left in a TCFZ. As part of the TCP process, Section 4.7.2 of OP-CL-201-009 required the site Fire Marshall, or designee, to confirm that a TCP was required and determine if the condition would be acceptable. Attachment 4 to OP-CL-201-009 outlined the criteria established by EC 0000376171 as being considered a negligible quantity. With these criteria established within OP-CL-201-009, the inspectors noted that a TCP for material within a TCFZ without appropriate compensatory measures could be approved by the Fire Marshall, or designee, because the quantity of material would be deemed to be negligible.

The Clinton Power Station Unit 1 Operating License (NPF-62), Condition 2.F permitted the licensee to make changes to the approved Fire Protection Program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. The inspectors determined that EC 0000376171 and OP-CL-201-009 could be used by fire protection personnel to justify acceptability of a quantity of material within a TCFZ that could challenge the ability to achieve and maintain safe shutdown in the event of a fire. As such, the inspectors determined that EC 0000376171 and OP-CL-201-009 were contrary to License Condition 2.F. However, the inspectors did not identify any instances where an inappropriate quantity of material to be left in a TCFZ had actually been approved on a TCP. Additionally, licensee site management had vigorously reinforced the expectation that no transient combustible materials were permitted in TCFZs. Therefore, the inspectors concluded that this issue constituted a violation of minor significance that is not subject to enforcement action in accordance with the NRC's Enforcement Policy. The licensee entered this issue into its corrective action program as AR 01023313 and changed OP-CL-201-009 to address the inspectors' concerns.

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

.1 (Closed) LER 05000461/2009-005-00, "Manual Scram on High Water Level Due to Reactor Recirc [Recirculation] Pump Trip."

On October 15, 2009, Unit 1 was manually scrammed following an unexpected trip of the 'B' reactor recirculation pump. Operators manually scrammed the reactor just before reactor vessel level reached the Level 8 (high level) reactor scram setpoint. After the unit was shut down, the licensee identified that the pump motor had failed due to an internal electrical fault. The licensee reported this event as a condition that resulted in the manual actuation of the reactor protection system in accordance with 10 CFR 50.73(a)(2)(iv)(A). The performance issues related to this event are discussed in Section 1R12.b.(1) of this inspection report. The inspectors concluded that the licensee's failure to correct a non-conforming condition with inadequate feed water level control system (FWLCS) response that resulted in a second reactor scram for the same cause was a finding of very low safety significance. LER 05000461/2009-005-00 is closed.

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

.2 Response to a Fire in the Turbine Building North Condenser Waterbox Area

a. Inspection Scope

On January 28, 2010, a fire was reported in the North Condenser Waterbox Area of the Turbine Building. Control Room operators promptly dispatched the plant fire brigade and evacuated the Turbine Building. The fire was extinguished within nine minutes and there was no significant damage. The inspectors reviewed the licensee’s response to the fire including fire brigade actions, evaluation of the cause, and extent of damage. The inspectors verified that operator response was in accordance with plant procedures and that equipment important to safety was not affected.

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

b. Findings

No findings of significance were identified.

4OA5 Other Activities

.1 Re-evaluation of a Previous Cross-Cutting Aspect

On January 14, 2010, Mr. F. A. Kearney, Exelon Generation Company, LLC (Exelon), Clinton Power Station, provided a response to an NRC Inspection Report issued on December 14, 2009, concerning activities conducted at Clinton Power Station. Specifically, the licensee disagreed with the cross-cutting aspect associated with a Non-Cited Violation identified in the inspection report, NCV 05000461/2009502-01, regarding the failure to obtain prior NRC approval for a change made to its Emergency Plan that decreased the effectiveness of the Plan. In the inspection report, the NRC determined that the cause for this finding was related to the cross-cutting area of Human Performance and its associated component for Decision Making (H.1(b)), because licensee personnel did not recognize that the removal of minimum on-shift emergency response staffing decreased the effectiveness of the Emergency Plan. The licensee indicated that this cross-cutting aspect was not reflective of present performance and that the finding did not have a cross-cutting aspect.

The licensee stated that the decision that resulted in the violation occurred in 2002, and was not reflective of present performance. Therefore, the licensee disagreed with the characterization of the cross-cutting aspect being classified as Human Performance with an associated component of Decision Making (H.1(b)).

The NRC staff reviewed the licensee's disagreement with the identification of the cross-cutting aspect associated with the Non-Cited Violation. After careful consideration, and review of the new revision of the NRC IMC 0612 guidance for assessing cross-cutting aspects, the NRC staff concluded that no cross-cutting aspects were associated with NCV 05000461/2009502-01. This matter is considered closed.

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

a. Inspection Scope and Documentation

On January 26, 2010, the inspectors conducted a walkdown of normally inaccessible portions of HPCS system piping in sufficient detail to reasonably assure the acceptability of the licensee's walkdowns (TI 2515/177, Section 04.02.d). The inspectors also verified that the information obtained during the licensee's walkdown was consistent with the items identified during the inspector's independent walkdown (TI 2515/177, Section 04.02.c.3).

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

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

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

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

b. Findings

No findings of significance were identified.

4OA6 Management Meetings

.1 Resident Inspectors' Exit Meeting

The inspectors presented the inspection results to Mr. F. Kearney and other members of the licensee's staff at the conclusion of the inspection on April 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:

- Removal of the cross-cutting aspect associated with Emergency Preparedness NCV 05000461/2009502-01, which was conducted with Mr. J. Peterson via telephone on February 1, 2010.

- The results of the Radiological Hazard Assessment and Exposure Controls and ALARA Controls inspection with Mr. F. Kearney and other members of the licensee’s staff on January 22, 2010. The inspectors confirmed that none of the potential report input discussed was considered proprietary.

- The results of the Inservice Inspection Activities Inspection with Mr. F. Kearney and other members of the licensee’s staff on January 22, 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.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

M. Baig, Engineering Programs, ISI
K. Baker, Design Engineering Senior Manager
S. Clary, Engineering Programs Manager
T. Chalmers, Operations Director
J. Cunningham, Operations Manager Developmental
J. Cummings, Welding Engineer
B. Davis, Acting Engineering Director
H. Do, Exelon Corporate ISI
J. Domitrovich, Work Management Director
C. Dunn, Shift Operations Superintendent
R. Frantz, Regulatory Assurance
M. Heger, Mechanical/Structural Design Engineering Manager
N. Hightower, Radiation Protection Operations Manager
S. Lakebrink, Design Engineering
K. Leffel, Operations Support Manager
M. Kanavos, Plant Manager
F. Kearney, Site Vice President
D. Kemper, Regulatory Assurance Manager
S. Kowalski, Engineering Response Manager
J. Peterson, Regulatory Assurance
J. Stovall, Radiation Protection Manager
J. Ufert, Fire Marshall
C. VanDenburgh, Nuclear Oversight Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000461/2010002-01	NCV	Failure to Control Transient Combustible Materials in Accordance with Fire Protection Program (Section 1R05.1.b.(1))
05000461/2010002-02	NCV	Failure to Control Combustible Gas Cylinders in Accordance with Fire Protection Program (Section 1R05.1.b.(2))
05000461/2010002-03	NCV	Interconnecting Floor Drains Between the Residual Heat Removal 'A' Pump Room and Radwaste Pipe Tunnel (Section 1R06.1.b.(1))
05000461/2010002-04	NCV	Failure to Recognize Examination Limitations for a Containment Penetration Weld (Section 1R08.1.b.(1))
05000461/2010002-05	FIN	Failure to Correct Inadequate FWLCS Response Resulted in High Reactor Vessel Water Level (Level 8) Scram (Section 1R12.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))

05000461/2010002-07	URI	Main Turbine Trip During On-line Testing (Section 1R20.2)
05000461/2010002-08	NCV	Inadequate Test Criteria in Standby Gas Treatment System Flow/Heater Operability Surveillance Test (Section 1R22.b.(1))
05000461/2010002-09	URI	Individual Contaminated in RT Hold Pump Room (Section 2RS2.3)

Closed

05000461/2010002-01	NCV	Failure to Control Transient Combustible Materials in Accordance with Fire Protection Program (Section 1R05.1.b.(1))
05000461/2009005-01	URI	Failure to Control Transient Combustible Materials in Accordance with Fire Protection Program (Section 1R05.1.b.(1))
05000461/2010002-02	NCV	Failure to Control Combustible Gas Cylinders in Accordance with Fire Protection Program (Section 1R05.1.b.(2))
05000461/2010002-03	NCV	Interconnecting Floor Drains Between the Residual Heat Removal ‘A’ Pump Room and Radwaste Pipe Tunnel (Section 1R06.1.b.(1))
05000461/2010002-04	NCV	Failure to Recognize Examination Limitations for a Containment Penetration Weld (Section 1R08.1.b.(1))
05000461/2010002-05	FIN	Failure to Correct Inadequate FWLCS Response Resulted in High Reactor Vessel Water Level (Level 8) Scram (Section 1R12.b.(1))
05000461/2010002-08	NCV	Inadequate Test Criteria in Standby Gas Treatment System Flow/Heater Operability Surveillance Test (Section 1R22.b.(1))
05000461/2009-005-00	LER	Manual Scram on High Water Level Due to Reactor Recirculation Pump Trip (Section 4OA3.1)

Discussed

05000461/2009004-01	URI	Interconnecting Floor Drains Between the Residual Heat Removal ‘A’ Pump Room and Radwaste Pipe Tunnel (Sections 1R06.1 and 4OA1.5)
05000461/2010-001-00	LER	Unanalyzed Leakage Pathway Affecting Residual Heat Removal A Pump Room Flooding Analysis (Section 1R06.1.b.(1))
05000461/2008004-01	FIN	Failure to Perform Adequate Post-Maintenance Testing Resulted in High Reactor Vessel Water Level (Level 8) Scram (Section 1R12.b.(1))
05000461/2009-005-00	LER	Manual Scram on High Water Level Due to Reactor Recirculation Pump Trip (Section 1R12.b.(1))
05000461/2009502-01	NCV	Implementation of a Change Which Decreased the Effectiveness of the Emergency Plan (Section 4OA5.1)
TI 2515/177	TI	Managing Gas Accumulation in Emergency Core Cooling, Decay Heat Removal, and Containment Spray Systems (Sections 1R04.3, 1R18.2, and 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.

1R04 Equipment Alignment

- CPS 3317.01, “Fuel Pool Cooling and Cleanup (FC),” Revision 23b
- CPS 3317.01E001, “Fuel Pool Cooling and Cleanup Electrical Lineup,” Revision 13
- CPS 3317.01V001, “Fuel Pool Cooling and Cleanup Valve Lineup,” Revision 12
- M05-1037, “P&ID Fuel Pool Cooling & Cleanup (FC),” Sheet 1, Revision W
- M05-1037, “P&ID Fuel Pool Cooling & Cleanup (FC),” Sheet 2, Revision AD
- M05-1037, “P&ID Fuel Pool Cooling & Cleanup (FC),” Sheet 3, Revision Y
- M05-1037, “P&ID Fuel Pool Cooling & Cleanup (FC),” Sheet 4, Revision X
- CPS 3212.01, “Plant Service Water,” Revision 29b
- CPS 3212.01V001, “Plant Service Water Valve Lineup,” Revision 20b
- CPS 3212.01V002, “Plant Service Water Instrumentation Valve Lineup,” Revision 8
- CPS 3212.01E001, “Plant Service Water Electrical Lineup,” Revision 11
- M05-1056, “Plant Service Water,” Sheet 001, Revision AP
- M05-1056, “Plant Service Water,” Sheet 002, Revision AH
- M05-1056, “Plant Service Water,” Sheet 003, Revision AB
- AR 00947143, “1WS45AA: Degrading Trend of WS Pipe Wall Thickness”
- AR 00917585, “Drawings Do Not Match 1WS01PA Pump”
- AR 00863681, “1WS048C: Piping Erosion Downstream of Valve”
- AR 00890965, “WS Pump Repair – Failure to Obtain Satisfactory Results”
- EC 374713, “Service Water System Operation with Rebuilt Pump Having Lower Performance Parameters than Original Design” Revision 0
- AR 00927791, “Recommend ACE for 1WS01PA Repairs/Reassembly”
- AR 00928644, “WS Strainer A Drain Piping Vibrates/1WS048A Not Full Shut”
- AR 00953213, “Pipe 1WS11D Below Acceptance Criteria for Wall Thickness”
- AR 01008855, “1WS253C Difficult to Operate”
- AR 01006974, “Line Stop Equipment Installation Engineering Change”
- CPS 3309.01, “High Pressure Core Spray (HPCS),” Revision 16a
- CPS 3309.01V001, “High Pressure Core Spray Valve Lineup,” Revision 11b
- CPS 3309.01V002, “High Pressure Core Spray Instrument Valve Lineup,” Revision 9
- CPS 3309.01E001, “High Pressure Core Spray Electrical Lineup,” Revision 7
- IS-1074-S, “High Pressure Core Spray (HP) System,” Sheet 1, Revision C
- M07-1074, “High Pressure Core Spray (HP) Piping 2” & Under,” Sheet 1, Revision AD
- M10-9074, “High Pressure Core Spray System (HP),” Sheet 1, Revision C
- CPS 3312.01V001, “Residual Heat Removal Valve Lineup,” Revision 17
- CPS 3312.01V002, “Residual Heat Removal Instrument Valve Lineup,” Revision 9

1R05 Fire Protection

- 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 1893.04M703, “712’ Turbine: Condenser Prefire Plan,” Revision 5a
- CPS 1893.04M711, “737’-762’ Turbine: LP & HP Heaters/Coolers Prefire Plan,” Revision 4a
- CPS 1893.04M111, “737 Auxiliary Building: Containment Personnel Hatch Prefire Plan,” Revision 4
- CPS 1893.04M721, “762’ Turbine: LP Heaters, Main Steam Valves Prefire Plan,” Revision 4
- National Fire Protection Association Code, Section 10: “Standard for Portable Fire Extinguishers,” 1975 & 1981
- Clinton Power Station Fire Protection Program, “Code Conformance Evaluation with Completed Checklists,” Revision 20
- AR 00972704, “Combustible Liquids Found in Transient Combustible Free Zone by NRC Resident Inspector”
- AR 01018554, “NRC Question on Fire Extinguisher at 709’ Turbine”
- AR 01020177, “NRC Request for FP Information”
- AR 01013399, “C1R12 Lesson Learned NRC Resident Observation Regarding Fire Protection Behaviors”
- Quick Human Performance Investigation AR 01013399, “C1R12 Lesson Learned NRC Resident Observation Regarding Fire Protection Behaviors”
- Quick Human Performance Investigation AR 01010601, “IEMA Identified Issue – Transient Combustible Material Violation”

1R06 Flooding Protection Measures

- CPS 4304.01, “Flooding,” Revision 4e
- 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
- EC 377321, Minor Revision to Calculation 034845(EMD), “Piping Stress Analysis for Residual Heat Removal System 1RH-09,” Revision 2-F
- 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”

1R08 Inservice Inspection Activities

- AR 01020881, “NRC Observation of NDE Activities in C1R12,” January 25, 2010
- AR 01020871, “Potential NRC NCV for Weld Accessibility for Examination,” January 25, 2010
- AR 01017892, “Wording in WPS Could be Misleading,” January 18, 2010
- AR 01017159, “C1R12 Unacceptable Surface Indication,” January 15, 2010
- AR 00843966, “Safety Related ASME Section III, QA Level 1,” November 12, 2008

- AR 00825723, “Inadequate Documentation Peer Review,” October 2, 2008
- AR 00763530, “Welding Heat Input Data not in Work Order,” April 15, 2008
- AR 00751868, “ASME Code Parts UTC not Documented,” March 19, 2008
- AR 00741384, “Flange Base Metal Repair,” February 26, 2008
- AR 00729527, “CRD Leakage Identified in CR11 Pressure Test,” January 31, 2008
- AR 00728432, “ASME Code Part Accepted Without Documentation,” January 29, 2008
- AR 00727598, “Leakage Noticed from CRD at 48-25,” January 27, 2008
- AR 00725951, “MT Indication at Toe of Weld 1CS608A,” January 23, 2008
- Certified Material Test Report, “ARCOS 90S-B3, Lot No. XA8450,” November 11, 2005
- Certified Material Test Report, “Spool R12-RT-33-1, Lot No. 982820,” December 28, 2009
- Drawing No 105D5975, “Penetration Pipe Spool,” Revision 5
- Examination Summary Sheet, “1-MS-D-10,” January 22, 2010
- NDE Report No C1-010, “Elbow-to-Pipe Weld 1RH-9-13-7,” January 11, 2008
- NDE Report No. C1-042, “Safe End-to-Nozzle Weld N4D-W-1,” January 23, 2008
- NDE Report No. 02-052, “Pipe-to-Penetration Weld 1-MS-B-11,” April 12, 2002
- Post-Weld Heat Treatment Record, “Weld W-11, Work Order 01259085,” January 15, 2010
- Post-Weld Heat Treatment Record, “Weld W-11, Work Order 01259085,” January 15, 2010
- Procedure GE-PDI-UT-1, “PDI Generic Procedure for the Ultrasonic Examination of Ferritic Pipe Welds,” Revision 6
- Procedure GE-PDI-UT-2, “PDI Generic Procedure for the Ultrasonic Examination of Austenitic Pipe Welds,” Revision 4
- Procedure GE-PDI-UT-3, “PDI Generic Procedure for the Ultrasonic Thru Wall Sizing in Piping Welds,” Revision 2
- Procedure GE-PT-100, “Procedure for Liquid Penetrant Examination Using Fluorescent and Visible Dye Liquid Penetrant Inspection Methods,” Revision 5
- Procedure MT- EXLN-100V3, “Procedure for Magnetic Particle Examination (Dry Particle, Color Contrast or Wet Particle, Fluorescent),” Revision 0
- Procedure GE-MT-100, “Procedure for Magnetic Particle Examination (Dry Particle, Color Contrast or Wet Particle, Fluorescent),” Revision 6
- Procedure GEH-UT-716. “Procedure for the Examination of Reactor Pressure Vessel Welds from the Outside Surface with Microtomo In Accordance With Appendix VIII,” Revision 3
- Procedure Qualification Record, “M05253-256,” September 11, 1989
- Procedure Qualification Record, “5A-5A-TS-001,” October 8, 1999
- Procedure Qualification Record, “M05002,” July 16, 1996
- Procedure Qualification Record, “01-01-T-001,” July 16, 1996
- Procedure Qualification Record, “01-01-T-002,” June 29, 1987
- Procedure Qualification Record, “01-01-S-001,” June 29, 1987
- Procedure Qualification Record, “01-01-TS-001,” September 9, 1982
- Procedure Qualification Record, “01-01-TS-101,” July 17, 2002
- Radiographic Records, “Welds W-5,6,11 and14,” January 15, 2010
- Weld Data Sheet, “W-11,” January 14, 2010
- Weld Data Sheet, “W-14,” January 14, 2010
- WPS-5A-5A-TS-200, Revision 3
- WPS-01-5A-TS-200, Revision 0
- WPS-01-01-TS-200, Revision 3
- Welder Performance Qualification, “JSW 1093,” February 10, 2006
- Welder Performance Qualification, “JSW 1093,” January 4, 2010
- Welder Performance Qualification, “JKB 5939,” February 5, 2007
- Welder Performance Qualification, “JKB 5939,” January 4, 2010
- Welder Performance Qualification, “JH 8196,” March 15, 2007
- Welder Performance Qualification, “JH 8196,” January 4, 2010

- Welder Performance Qualification, “LSC 1182,” December 16, 2002
- Welder Performance Qualification, “LSC 1182,” January 4, 2010
- Welder Performance Qualification, “MSS 2663,” October 14, 2008
- Welder Performance Qualification, “MSS 2663,” January10, 2010

1R12 Maintenance Effectiveness

- ER-AA-310-1001, “Maintenance Rule – Scoping,” Revision 4
- Maintenance Rule (a)(1) Action Plan for Reliability Performance Criteria for Reactor Scrams Under Plant Level Performance Criteria (95-00-1) , November 19, 2009
- Maintenance Rule (a)(1) Determination for Reliability Performance Criteria 95-00-1, October 15, 2009
- AR 00983144, “FWLCS Resulted in Reactor Scram – RR B Pump Trip 10/15/09”
- AR 00979700, “RR B Trip – Resulting in Reactor Scram”
- AR 00991826, “OTDM Needed to Determine When to Perform FWLC Dynamic Tuning”
- AR 00807670, “NRC ID: No Evaluation for 2/08 Scram From 1 RR Pump Not Per USAR”
- LER 05000461/2009-005-00, “Manual Scram on High Water Level Due to Reactor Recirc [Recirculation] Pump Trip,” December 8, 2009
- Apparent Cause Evaluation AR 00983144, “Organizational Decision Making for Feed Water Level Control System (FWLCS),”
- Root Cause Evaluation AR 00979700, “1B33C001B: RR B Trip – Resulting in Reactor Scram,” November 9, 2009
- OTDM 991826-02, “Determine When to Perform the Feedwater Level Control (FWLC) System Dynamic Fine Tuning,” November 26, 2009
- OTDM 00807670-02, “Determine Whether to Continue Operation of the Plant With a Deficient Feedwater Level Control (FWLC) System Response or to Shutdown and Replace the FWLC System Relay, Circuit Card, and Perform FWLC System Dynamic Tuning,” October 21, 2008
- NNOE 1016173-16-01, “Reactor Recirculation Discharge Isolation Valve Would Not Close”
- CPS 3302.01, “Reactor Recirculation (RR),” Revision 31a
- AR 01020458, “Excessive Wear Found on Internals of Valve 1B33F067B”
- C1R12 Startup PORC – RR F067A/B White Paper, “Reactor Recirculation Discharge Valves”
- OTDM 10/16/2009 (IR#979732), “Reactor Recirculation (RR) Loop B Discharge Valve Failure to Close Electrically”
- AR 00979732, “Discharge Valve 1B33F067B Would Not Shut Following Pump Trip”
- OP-AA-108-108, “Unit Restart Review,” Revision 9
- Prompt Investigation 1016173, “1B33F067B – Discovered Cracked Limitorque Housing”
- AR 01015786, “1B33F067B Failed To Fully Stroke Closed”
- AR 01016173, “1B33F067B – Discovered Cracked Limitorque Housing”
- AR 01020466, “Minor Internal Wear Found in Valve 1B33F067A”
- EACE 1016173-07, “1B33F067B – Discovered Cracked Limitorque Housing”
- ER-AA-310, “Implementation of the Maintenance Rule,” Revision 8
- 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
- IR 00968545-04, “(a) (1) Action Plan Development and Action Plan (Monitoring) Goal Setting Template”
- IR 00968545, “MRULE (A) (1) Determination Requires Revision”
- IR 00904292, “NSPS DSC Card Maintenance Rule Functional Failure Review”
- IR 00944238, “Maintenance Rule Performance Criteria Exceeded for System”
- IR 00974262, “NSPS HPCS-1 Circuit Card Failed – Second Failure”

- GE Hitachi 22A7870, “Digital Signal Conditioner Card,” Revision 2
- GE 22A7854, “High Pressure Core Spray (HPCS) System Logic Cards,” Revision 3

1R13 Maintenance Risk Assessments and Emergent Work Control

- 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
- AR 01016188, “ERAT SVC Trip”
- AR 01016002, “0VV100PA ERAT SVC Cooling Skid Pump P1 Degrading”
- AR 00991384, “C1F51 LL – Evaluation of Forced Outage Shutdown Risk”
- AR 01023865, “Protected Systems Not Marked in the Plant”
- AR 01003968, “SDC Risk Review and Potential Equipment Vulnerabilities”
- AR 01007466, “SDC Risk Review and Potential Equipment Vulnerabilities – FC System”
- AR 01008974, “Non-Conservative Decision Making Supporting OPDRV Procedures”
- 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”
- Plant Operations Review Committee Meeting Minutes, Meeting Number 09-030, December 9, 2009
- 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
- EC 357294, “Temporary Configuration Change for CPS 8117.04,” Revision 0
- EC 376912, “Evaluate Hole Size for OPDRV,” Revision 1
- CPS 3711.01, “CPS Operations With The Potential To Drain The Reactor Vessel (OPDRV),” Revision 0
- 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

1R15 Operability Evaluations

- 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
- AR 00976295, "ECCS Floor Drain Piping Connected to the Radwaste Pipe Tunnel"
- AR 01017629, "Excess Flow Check Valve (1SM008) Slow to Reopen"
- AR 01008721, “9069.01 Flow Data Less Than Acceptable (Div 3 SX)”
- AR 00988981, “Potential Non-Conservative DO Tech Spec 3.8.3”
- AR 01032794, “2010 CDBI FASA – DG Fuel Oil Consumption Rate”

- AR 01035743, “Operability Evaluation Returned By SM With Comments”
- AR 01040035, “NOS ID Elevation For Operability Evaluations”
- CPS 9069.01, “Shutdown Service Water Operability Test,” Revision 46c
- CPS 9069.01, “Shutdown Service Water Operability Test,” Revision 46e
- CPS 9069.01D001, “SX System Operability Data Sheet”
- ECR 389858, April 9, 2009
- ECR 393299, December 22, 2009
- Operability Evaluation # 01032794-02, “1DG01KA – Division 1 Emergency Diesel Generator (EDG)”
- Calculation EAD-DG-01, Revision 2B
- Calculation 01DO06, “Diesel Fuel Oil Storage Requirements,” Revision 7
- Transmittal Of Design Information, CPS-07-005, “Div 1 DG Loading for Input to Calc 01DG06,” Revision 0
- Exelon Nuclear Procurement Engineering Standard PES-P-006, “Diesel Fuel Oil,” Revision 4
- Prompt Investigation “2010 CDBI FASA – DG Fuel Oil Consumption Rate,” IR 01032794
- Regulatory Guide 1.137, “Fuel Oil Systems For Standby Diesel Generators,” Revision 1
- ANSI/ANS-59.51-1997, “Fuel Oil Systems For Safety-Related Emergency Diesel Generators”

1R18 Plant Modifications

- EC 369979, "Line Stop on Line 1WS33BA-20" Cross Tie Between WS [Plant Service Water] and Div 1 SX [Shutdown Service Water] to Repair MOV [Motor Operated Valve] 1SX014A," Revision 2
- WO 01149327-01, “Modification of Line Stopple in Support of 1SX014A Replacement”
- EC 377772, "Jumper Out High Compressor Oil Temperature Trip on 1VP04CB," Revisions 0 and 1
- WO 1259730-20, “Install TMOD EC 377772 on 1VP04CB”
- AR 00838073, “Non-Conforming Shielding on Suppression Pool Clean-up & Transfer Pump ‘B’ Suction Piping”
- EC 371560, “Installation of High Point Vent on Line 1HP02A-14”,” Revision 1
- CPS 3309.01, “High Pressure Core Spray,” Revision 16a
- CPS 9051.05, “HPCS Discharge Header Filled and Flow Path Verification,” Revision 27c
- Work Order 01160626, “Add Vent Valves to Resolve Trapped Air Voids in Pipe,” March 16, 2009
- Work Order 01291459, “9051.05R20 OP HPCS Discharge Header Filled and Flow Path Verification,” December 22, 2009
- AR 00939342, “NOS ID HPCS Vent Install Schedule Not Per Commitment”
- AR 00956892, “NOS ID Enhancement HPCS Vent Line Modification EC Could Discuss Hydraulics”
- AR 01021283, “1E22F382: Incomplete Testing of EC 371560”

1R19 Post-Maintenance Testing

- WO 00918494-20, "1SX01PB: Perform SX Pump Operability Test"
- WO 00918494-10, "1SX01PB: PMT-Collect Vibration/Thermography"
- WO 00790605-03, “OP 1E21F006-Post-LLRT”
- CPS 9843.01, “ISI Category “A” Valve Leak Rate Test,” Revision 35
- CPS 9843.01D002, “Category A Valve Leak Rate Test Via Flowmeter,” Revision 24b
- CPS 9843.01V002, “Leak Rate Testing of LPCS Injection,” Revision 27b
- AR 01015235, “1E21F006 (LPCS Check Valve) Failed to Seat – Test Failure”
- AR 01016798, “9843.01D002 Error in Corrected Pressure Calculation”

- WO 00750261-04, “OP 1E12F025A: As-Left LLRT (9861.02D018 Special Test)”
- CPS 9861.02D018, “LLRT Data Sheet for Retest/Pretest/Special Test,” Revision 27a
- CPS 9861.05, “Water Local Leak Rate Testing,” Revision 24e
- CPS 9069.01, “Shutdown Service Water Operability Test,” Revision 46e
- CPS 9069.01D001, “SX System Operability Data Sheet,” Revision 44
- AR 01021145, “Breakers Not Listed on 3514.01E006
- WO 00002664-15, “Rework 1B21F022B In Event of LLRT Failure – PMT 9861.04D002 MSIV LLRT”
- WO 00002665-07, “Rework 1B21F022C In Event of LLRT Failure – PMT 9861.04D003 MSIV LLRT”
- WO 01303779-04, "1SM008: Leak Rate Test Retest”
- CPS 3302.01, “Reactor Recirculation (RR),” Revision 31a
- CPS 3515.01, “Operation of 6900/4160/480V Circuit Breakers,” Revision 5a
- CPS 8216.11, “Main Steam Isolation Valve Maintenance,” Revision 22e
- CPS 8410.07, “Reactor Recirculation 6900V Vacuum Circuit Breaker Maintenance,” Revision 11
- CPS 8410.21, “Westinghouse DHP 6900, 4160V Power Circuit Breaker,” Revision 6
- CPS 9067.01, “Standby Gas Treatment System Train Flow/Heater Operability,” Revision 31b
- CPS 9067.01D001, “SGTS Train Flow/Heater Operability Data Sheet,” Revision 27c
- CPS 9861.04D002, “MSIV B LLRT Data Sheet (1MC-8),” Revision 25d
- Work Order 01128244, “MC008 LLRT Requirements (MSIV – B) and PIT 1E32-F001E,” December 21, 2009
- Work Order 01130746, “Breaker Installation Support and Functional Testing of New Breaker Installed at 1RR02ED,” March 3, 2010
- Work Order 01308289, “9067.01A20 OP SGTS Train Flow/Heater Operability,” March 2, 2010
- Work Order 00002664, “Rework 1B21-F022B in Event of LLRT Failure” February 19, 2010
- Work Order 00002665, “Rework 1B21-F022C in Event of LLRT Failure” February 19, 2010
- Drawing E03-1RR02ED, Sheet 1 Revision K

1R20 Outage Activities

- OP-AA-108-108, “Unit Restart Review,” Revision 9
- CPS 3001.01C001, “Preparation for Startup Checklist,” Revision 17c
- OP-AA-108-108, “Unit Restart Review,” Revision 9
- CPS 3007.01, “Preparation and Recovery from Refueling Operations,” Revision 14d
- MA-AA-716-008, “Foreign Material Exclusion Program,” Revision 4
- ER-AA-600-1023, “ORAM-Sentinel and Paragon Model Capability,” Revision 4
- OP-AA-108-107, “Switchyard Control,” Revision 2
- OP-AA-108-110, “Evaluation of Special Tests or Evolutions,” Revision 2
- OP-CL-108-107-1001, “Interface Between AmerinIP and Clinton Power Station for Switchyard Operations, Maintenance, and Engineering,” Revision 10
- OU-AA-103, “Shutdown Safety Management Program,” Revision 9
- OU-CL-104, “Shutdown Safety Management Program Clinton Power Station,” Revision 4
- CPS 1401.09, “Control of System and Equipment Status,” Revision 7
- CPS 4006.01, “Loss of Shutdown Cooling,” Revision 4d
- CPS 3312.02, “Alternate Shutdown Cooling (A-SDC) Methods,” Revision 9
- CPS 3312.03, “RHR – Shutdown Cooling (SDC) & Fuel Pool Cooling and Assist (FPC&A),” Revision 6b
- CPS 9000.06, “Reactor Coolant and Vessel Metal/Pressure/Temperature Limit Logs,” Revision 31b

- CPS 9000.06D001, “Heatup/Cooldown, Inservice Leak and Hydrostatic Testing 30 Minute Temperature Log,” Revision 30a
- AR 01020111, “NRC Observation of Scaffold West Side 2B LP Heater”
- AR 01026020, “NRC – T-Bolts Not Fastened for Shield Storage Box in Drywell”
- AR 01038847, “Regulator Questions Assumptions Used in Calculation”
- Engineering Evaluation EE 00-073, “Storage of Boxes of Lead Shielding Inside Drywell,” Revision 2
- CPS 3001.01, “Preparation For Startup and Approach to Critical
- CPS 3001.01C002, “Mode 2 Checklist,” Revision 16a
- CPS 3021.01, “Drywell Close Out (Long Form),” Revision 13b
- CPS 3312.03, “RHR – Shutdown Cooling (SDC) and Fuel Pool Cooling and Assist (FPC&A),” Revision 6
- CPS 3312.03C001, “Alternate Shutdown Cooling Temperature Monitoring Checklist,” Revision 0a
- CPS 9000.06, “Reactor Coolant and Vessel Metal/Pressure/Temperature Limit Logs,” Revision 31b
- CPS 9000.06D001, “Heatup/Cooldown, Inservice Leak and Hydrostatic Testing 30 Minute Temperature Log,” Revision 30a
- CPS 9000.06D002, “Vessel Head and Shell Flange Temperature log,” Revision 30
- CPS 9000.06D003, “Shutdown Cooling Temperature Data Sheet,” Revision 30d

<u>1R22 Surveillance Testing</u>

- Clinton Power Station Technical Specifications
- Clinton Power Station Updated Safety Analysis Report, Revision 13
- CPS 9067.01, “Standby Gas Treatment System Train Flow/Heater Operability,” Revision 31a
- CPS 9067.01D001, “SGTS Train Flow/Heater Operability Data Sheet,” Revision 27c
- AR 01007524, “NRC Questions Regarding VG Surveillance Testing”
- WO 01128241-01, “OP MC005 *LLRT Requirements (MSIV – C) and 1E32-F001J PIT”
- WO 01128242-01, “OP MC006 *LLRT Requirements (MSIV – A) and PIT 1E32-F001A”
- WO 01128243-01, “OP MC007 *LLRT Requirements (MSIV – D) and PIT 1E32-F001N”
- WO 01128244-01, “OP MC008 *LLRT Requirements (MSIV – B) and PIT 1E32-F001E”
- AR 01017464, “9861.04 LLRT on MSL A, B, and C Test Failure”
- CPS 9861.04, “MSIV Local Leak Rate Test (MC-5,6,7,8),” Revision 26
- CPS 9861.04D001, “MSIV A LLRT Data Sheet (1MC-6),” Revision 25d
- CPS 9861.04D002, “MSIV B LLRT Data Sheet (1MC-8),” Revision 25d
- CPS 9861.04D003, “MSIV C LLRT Data Sheet (1MC-5),” Revision 25e
- CPS 9861.04D004, “MSIV D LLRT Data Sheet (1MC-7),” Revision 25d
- WO 01141697-01, “Perform 9080.22 DG 1B Integrated Test (All Sections)”
- CPS 9080.22, “Diesel Generator 1B – ECCS Integrated,” Revision 31
- WO 01112688-01, “EM 1E12-F042A Thrust Verification”
- MA-AA-723-300, “Diagnostic Testing of Motor Operated Valves,” Revision 3
- MA-AA-723-300-1004, “QUIKLOOK Diagnostic Test Equipment / Sensor Guideline,” Revision 2
- AR 01017629, “Excess Flow Check Valve (1SM008) Slow to Reopen”
- AR 01023711, “Excess Flow Check Valve 1SM008 Slow to Re-open Restoration”
- AR 01023029, “NRC Question on Excess Flow Check Valves”
- AR 01023707, “Excess Flow Check Valve Testing and Restoration Enhancement”
- CPS 9864.01, “Excess Flow Check Valve Operability Test,” Revision 37a
- CPS 9864.01D004, “Low Pressure Excess Flow Check Valve Test Data Sheet for 1SM008,” Revision 29b

- IST-CPS-BDOC-V-02, “CPS Inservice Testing Bases Document,” Revision August 10, 2009
- WO 01274550-01, “Excess Flow Check Valve Open/Close Exercise (1SM008)”
- WO 01274799-01, “Excess Flow Check Valve 1SM008 PIT Test”
- CPS 9053.06, “Containment Spray System Functional Test,” Revision 32
- AR 01032431, “NRC Questions on 9080.22”
- AR 01040653, “Possible UFSAR Typo for VC System”
- AR 01039854, “Enhancement to 9080.22 Surveillance Requirement Reference”
- WO 01141699-01, “OP 9080.27 Unit Power Supply Normal Transfer Operability”
- CPS 9080.27, “Unit Power Supply Manual Transfer Operability,” Revision 1
- WO 01128963-01, “9080.19 OP DG 1B Overcrank Delay, Overcurrent, Trip Bypass Test, and Trip Bypass Operability”
- CPS 9080.19, “DG 1B Overcrank Delay, Overcurrent, Trip Bypass Test, and Trip Bypass Operability,” Revision 0c
- AR 01032998, “Incorrect Non-Technical Specification Data Logged for 9080.22”
- EC 378623, “Evaluate 9080.22 Surveillance Results for Time to Start RHR B Motor,” Revision 0
- WO 01150602-01, “OP 9059.01R20 Leak Rate Test Vessel Pressure Test”
- CPS 9059.01, “Reactor Coolant System Leakage Test,” Revision 7c
- WO 00918186, “Perform 9861.05D001 LPCS Water Test”
- CPS 9861.05D001, “RHR A/LPCS Water Leak Rate Test Data Sheet (S-MC021K01 & S-MC038K04),” Revision 24c
- CPS 2761.02, “Leak Rate Testing Equipment Operation,” Revision 5b
- CPS 3506.01C003, “Diesel Generator 1C Pre-Start Checklist,” Revision 3f
- CPS 3506.01C005, “Diesel Generator Start Log,” Revision 1
- CPS 3506.01D003, “Diesel Generator 1C Operating Logs,” Revision 3
- CPS 9843.01, “ISI Category ‘A’ Valve Leak Rate Test,” Revision 35
- CPS 9861.02, “LLRT Requirements and Type ‘C’ (Air) Local Leak Rate Testing,” Revision 42e
- CPS 9861.02D014, “LLRT Data Sheet for 1MC036 – LPCS Injection,” Revision 27
- CPS 9080.23, “Diesel Generator 1C – ECCS Integrated,” Revision 30a
- CPS 9080.23D001, “DG 1C – ECCS Integrated Data Sheet,” Revision 24e
- CPS 9080.23E001, “DG 1C – ECCS Integrated Electrical Lineup,” Revision 21b
- Work Order 01136235, “9080.23R20 OP *DG 1C Integrated Test,” January 21, 2010
- Work Order 01144801, “9843.01 *03 LRT* Category ‘A’ Valve Leak Rate Test (1E21-F005) Low Pressure Core Spray Injection,” January 31, 2010
- Work Order 01299837, “9080.23A23 OP DG 1C Operability – Monthly Test,” January 25, 2010
- AR 01015796, “C1R12 LL MC036 LLRT LPCS Injection Required More Time”
- AR 01020527, “NOS ID MSIV LLRT Data Anomalies”
- AR 01032998, “Incorrect Non-Technical Specification Data Logged for 9080.22”
- WO 01299873-01, “OP Fuel Pool Cooling Pump 1B and 1A Valve IST Testing”
- CPS 9061.10, “Fuel Pool Cooling Valve Operability,” Revision 44
- Drawing M05-1037, “P&ID Fuel Pool Cooling & Clean Up (FC),” Sheet 1, Revision W
- Drawing M05-1037, “P&ID Fuel Pool Cooling & Clean Up (FC),” Sheet 2, Revision AD
- Drawing M05-1037, “P&ID Fuel Pool Cooling & Clean Up (FC),” Sheet 3, Revision Y
- Drawing M05-1037, “P&ID Fuel Pool Cooling & Clean Up (FC),” Sheet 4, Revision X
- IR 01047797, “Failed to Reset Stopwatch During SX-C Surveillance”
- IR 01047812, “Backflow Through 1SX001C Noted During SX Pump C Surveillance”
- WO 01297427-01, “OP 9059.01 SX Pump ‘C’ Oper Test”
- CPS 9069.01, “Shutdown Service Water Operability Test,” Revision 46e

1EP4 Emergency Action Level and Emergency Plan Changes

- Letter from Mr. F. A. Kearney, Exelon Generation Company, LLC (Exelon), Clinton Power Station dated January 14, 2010

2RS1 Radiological Hazard Assessment and Exposure Controls

- RP-AA-462, "Controls for Radiographic Operations," Revision 7
- RP-AA-460, "Controls for High Radiation and Locked High Radiation Areas," Revision 19
- RP-AA-460-002, "Additional High Radiation Exposure Control," Revision 0
- RWP and Associated ALARA File, "RWP 10010079; C1R12 Drywell – ISI Inside Bio-shield," Revision 0
- RWP and Associated ALARA File, "RWP 10010087; C1R12 Drywell – Scaffolding," Revision 0
- RWP and Associated ALARA File, "RWP 10010088; C1R12 Drywell Shielding," Revision 0
- RWP and Associated ALARA File, "RWP 10010098; C1R12 Drywell – SRV Work," Revision 0
- RWP and Associated ALARA File, "RWP 10010101; RT Pipe Replacement," Revision 0
- RWP and Associated ALARA File, "RWP 10010150; C1R11 [sic] Refuel Floor Work (No Cavity)," Revision 0

2RS2 Occupational ALARA Planning and Controls

- Prompt Investigation 1017853, "Individual Contaminated in RT Hold Pump Room," date not provided
- RP-CL-1004, "Radiological Engineering/ALARA Group "How To" for ALARA Planning," Revision 4
- RP-AA-4003, "Guidelines for Daily Radiation Protection Outage Report," Revision 2
- RP-AB-3001, "BRAC Point Radiation Surveys," Revision 0

4OA1 Performance Indicator Verification

- Nuclear Energy Institute 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 6
- AR 00976295, "ECCS Room Floor Drain Piping Connected to RW Pipe Tunnel"

4OA2 Identification and Resolution of Problems

- OP-AA-201-009, "Control of Transient Combustible Material," Revision 10
- OP-CL-201-009, "Control of Transient Combustible Material," Revision 0
- EC 0000376171, "Clinton Power Station (CPS) Fire Protection Program: Evaluation of Transient Combustible Free Fire Zones (TCFZs) and Determination of Negligible Quantities of Combustible Materials and Their Impact on TCFZs," Revision 0
- AR 01023313, "NRC Concern About Fire Protection EC 376171 on TCFZ"

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

- LER 05000461/2009-005-00, “Manual Scram on High Water Level Due to Reactor Recirc [Recirculation] Pump Trip,” December 8, 2009

4OA5 Other Activities

- Exelon Letter RS-09-173, “Response to Request for Additional Information Regarding Generic Letter 2008-01,” December 15, 2009
- Exelon Letter, “Generic Letter 2008-01 Nine-Month Response for Clinton Power Station,” January 11, 2008
- CPS 9052.04, “LPCS/RHR ‘A’ Discharge Header Filled and Flow Path Verification,” Revision 27d
- EC 369907, “NRC Generic Letter 2008-01 System Scope and SER 02-05,” Revision 1
- EC 371529, “Generic Letter 2008-01 System Evaluation Template – HPCS Evaluation,” Revision 1
- EC 371530, “Generic Letter 2008-01 System Evaluation Template – LPCS Evaluation,” Revision 1
- EC 371531, “Generic Letter 2008-01 System Evaluation Template – RHR Evaluation,” Revision 1
- EC 371609, “Generic Letter 2008-01 UT Testing Acceptance Criteria Div 1,” Revision 1
- EC 371660, “Generic Letter 2008-01 UT Testing Acceptance Criteria Div 3,” Revision 1
- EC 377499, “SER 2-05 Rev. 1 System Evaluation Template SLC Evaluation,” Revision 0
- Work Order 01152279, “NRC GL 2008-01 ECCS Gas Intrusion Field Activities,” January 21, 2009
- Work Order 01298961, “EP Perform UT Testing to Check for Accumulated Air (HPCS),” February 3, 2010
- Work Order 01303658, “EP Perform UT Testing to Check for Accumulated Air (LPCI A),” January 21, 2010
- AR 00728092, “NRC Generic Letter 2008-01 Managing Gas Accumulation”
- AR 00790817, “NRC GL 2008-01 ECCS Gas Intrusion Field Activities”
- AR 00828402, “Implement GL 2008-01 Commitments”
- AR 00958740, “SER 02-05 Review RCIC System Enhancements”
- AR 00988801, “NRC GL 2008-01 Request for Additional Information”
- AR 01017789, “Air Voids Discovered During UT Examinations of ‘A’ RHR”
- AR 01022886, “RHR ‘C’ Pump Suction Voiding”
- AR 01036811, “LPCI ‘A’ 9052.04 Scheduled Prior to Engineering UT”

LIST OF ACRONYMS USED

ADAMS	Agency-wide Documents and Management System
ASME	American Society of Mechanical Engineers
ALARA	As-Low-As-Is-Reasonably-Achievable
AR	Action Request
BI	Barrier Integrity
BTP	Branch Technical Position
BTU	British Thermal Unit
BWR	Boiling Water Reactor
CCDP	Conditional Core Damage Probability
ΔCDF	Change in Core Damage Frequency
CFR	Code of Federal Regulations
CNO	Chief Nuclear Officer
CPS	Clinton Power Station
ºF	Degrees Fahrenheit
EC	Engineering Change
ECCS	Emergency Core Cooling System
EPD	Electronic Personal Dosimeter
EOP	Emergency Operating Procedure
FDT	Fire Dynamics Tools
FT^2	Square Feet
FWLCS	Feedwater Level Control System
FIN	Finding
GDC	General Design Criteria
GPM	Gallons-Per-Minute
HEPA	High Efficiency Particulate Air
HP	Health Physics
HPCS	High Pressure Core Spray
HRA	High Radiation Area
IMC	Inspection Manual Chapter
IE	Initiating Events
IP	Inspection Procedure
ISI	Inservice Inspection
IST	Inservice Testing
LER	Licensee Event Report
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MOV	Motor Operated Valve
MS	Mitigating Systems
MSIV	Main Steam Isolation Valve
MT	Magnetic Particle Examination
NCV	Non-Cited Violation
NDE	Nondestructive Examination
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
OTDM	Operations Technical Decision Making
OPDRV	Operations with the Potential to Drain the Reactor Vessel

PARS	Publicly Available Records
P&IDs	Piping and Instrumentation Diagrams
PRA	Probalistic Risk Assessment
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPM	Radiation Protection Manager
RPT	Radiation Protection Technician
RT	Reactor Water Clean-up
RWP	Radiation Work Permit
SDP	Significance Determination Process
SGT	Standby Gas Treatment
SPAR	Standardized Plant Analysis Risk
SR	Surveillance Requirement
SRA	Senior Reactor Analyst
SRV	Safety Relief Valve
SSC	Structures, Systems, and Components
SX	Shutdown Service Water
TCFZ	Transient Combustible Free Zone
TCP	Transient Combustible Permit
TI	Temporary Instruction
TS	Technical Specification
URI	Unresolved Item
UFSAR	Updated Final Safety Analysis Report
UT	Ultrasonic Examination
VAR	Volt-Ampere Reactive
VG	Standby Gas Treatment
VHRA	Very High Radiation Area
VP/WO	Drywell Cooling/Chilled Water
WO	Work Order
WR	Work Request
WS	Plant Service Water

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Sincerely,

/RA/

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

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License No. NPF-62

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SUBJECT: CLINTON POWER STATION NRC INTEGRATED INSPECTION REPORT 05000461/2010-002

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