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

May 8, 2012

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

SUBJECT: CLINTON POWER STATION - NRC INTEGRATED INSPECTION REPORT 05000461/2012-002

Dear Mr. Pacilio:

On March 31, 2012, 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 19, 2012, with Mr. W. Noll 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 NRC-identified finding and one self-revealed finding of very low safety significance were identified. Each of these findings was determined to involve a violation of NRC requirements. Additionally, one licensee-identified violation, which was determined to be of very low safety significance, was reviewed by the inspectors and is listed in this report.

Because of the very low safety significance and because they were entered into your corrective action program, the NRC is treating the above inspector-identified, self-revealed, and licensee-identified violations as non-cited violations (NCVs) consistent with Section 2.3.2 of the NRC Enforcement Policy. If you contest any NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001; with copies to the Regional Administrator, Region III; the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at Clinton Power Station. In addition, if you disagree with a cross-cutting aspect assignment in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement to the Regional Administrator, Region III, and the NRC Resident Inspector at Clinton Power Station.

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

Sincerely,

/RA/

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

Docket No. 50-461
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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	05000461/2012-002
Licensee:	Exelon Generation Company, LLC
Facility:	Clinton Power Station, Unit 1
Location:	Clinton, IL
Dates:	January 1 through March 31, 2012
Inspectors:	B. Kemker, Senior Resident Inspector D. Lords, Resident Inspector S. Bell, Health Physicist J. Bozga, Reactor Inspector C. Brown, Reactor Inspector J. Cassidy, Senior Health Physicist R. Orlikowski, Project Engineer D. Szwarc, Reactor Inspector S. Mischke, Resident Inspector, Illinois Emergency Management Agency
Approved by:	M. Ring, Chief Branch 1 Division of Reactor Projects

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SUMMARY OF FINDINGS

IR 05000461/2012-002, 01/01/12 – 03/31/12; Clinton Power Station, Unit 1; Maintenance Effectiveness, Identification and Resolution of Problems.

This report covers a three-month period of inspection by the resident inspectors and announced baseline inspections by regional inspectors. Two Green findings, each 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. A self-revealed finding of very low safety significance was identified with an associated non-cited violation of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The licensee failed to incorporate operating experience into its preventive maintenance practices associated with steam bypass system control circuit cards. Specifically, during two operating experience driven initiatives performed by the licensee in 2001 and 2007, and once again on September 24, 2011, the licensee failed to implement any preventive maintenance activity for critical component circuit cards, which resulted in age-related failure and a reactor scram on November 29, 2011. The licensee initiated corrective actions to replace system circuit cards, perform periodic replacement/refurbishment maintenance activities, and trend circuit card performance during routine calibration.

  The finding was of more than minor significance because it was sufficiently similar to IMC 0612, "Power Reactor Inspection Reports," Appendix E, "Examples of Minor Issues," Example 7(c), in that this violation of 10 CFR 50.65(a)(3) had a consequence "…such as equipment problems attributable to failure to take industry operating experience into account when practicable." The finding was a licensee performance deficiency of very low safety significance because it: (1) did not contribute to the likelihood of a loss-of-coolant accident initiator, (2) did not contribute to both the likelihood of a reactor scram 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 concluded that this finding affected the cross-cutting area of human performance. Specifically, in the area of work control, the licensee did not appropriately coordinate work activities by incorporating actions to plan work activities to support long-term equipment reliability by scheduling maintenance as more preventive than reactive. (IMC 0310 H.3(b)) (Section 4OA2.3.b.2)

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." The licensee failed to establish an adequate procedure to perform required leak rate testing for the Low Pressure Coolant Injection from Residual

Heat Removal 'A' Check Valve. Specifically, the surveillance test procedure resulted in unacceptable preconditioning of the valve prior to a leak rate test measurement due to improper test sequencing. In addition, the licensee failed to correctly evaluate a failed leak rate test of the valve. The licensee entered this issue into its corrective action program for evaluation and initiated corrective actions to revise the test procedure and train engineering personnel.

The finding was of more than minor significance since it was associated with the Procedure Quality attribute for the containment 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. Because the preconditioning altered the as-found condition of the check valve, the data collected through the performance of the surveillance test was not fully indicative of the true valve performance trend. Additionally, the licensee's failure to correctly evaluate the initial failed leak rate test would become a more significant safety concern if left uncorrected because it could reasonably result in an unrecognized condition with a check valve failing to fulfill a safety-related function. Therefore, this performance deficiency had a direct effect on the licensee's ability to fully assess the past operability, as well as the ability to trend as-found data for the purpose of assessing the reliability of the check valve. The finding was a licensee performance deficiency of very low safety significance because it would not result in exceeding the Technical Specification limit for reactor coolant system leakage and would not have likely affected mitigation systems resulting in a loss of safety function. In addition, the finding did not represent an actual open pathway in the physical integrity of the reactor containment. Based on consultation and review with the Regional Senior Reactor Analyst, the inspectors concluded that the finding did not result in an increase in the likelihood of an initiating event such as an inter-system loss-of-coolant accident or a containment bypass event because the redundant isolation valve and closed loop system piping passed leak rate measurement testing during the refueling outage with considerable margin. The inspectors concluded that this finding affected the cross-cutting area of human performance. Specifically, the licensee did not have adequately trained and knowledgeable personnel available to correctly evaluate the cause of the initial failed leak rate measurement test and to ensure that appropriate actions to correct the test sequence in the procedure were identified. (IMC 0310, H.2(b)) (Section 1R12.b.1)

B. Licensee-Identified Violations

A violation of very low safety significance that was identified by the licensee has been reviewed by the inspectors. Corrective actions planned or taken by the licensee have been entered into the licensee's corrective action program.

REPORT DETAILS

Summary of Plant Status

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

- On February 4th, the licensee reduced power to about 90% to attempt recovery of a control rod that would not couple during plant start up from the refueling outage in December 2011. The unit was returned to full power the same day.
- On March 4th, the licensee reduced power to about 72% to perform control rod sequence exchange and main turbine control/stop/intermediate valve and main steam isolation valve testing. The unit was returned to full power the same day.

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignment (71111.04)

.1 Quarterly Partial System Walkdowns (71111.04Q)

a. Inspection Scope

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

- Division 1 Diesel Generator (DG) during maintenance on the Division 2 DG;
- Drywell Purge Train ‘A’ during maintenance on Drywell Purge Train ‘B’; and
- Residual Heat Removal (RHR) Train ‘A’ during scheduled maintenance of 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.

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

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors performed a complete system alignment inspection of the Standby Liquid Control System to verify the functional capability of the system. This system was selected because it was considered both safety significant and risk significant in the licensee's probabilistic risk assessment. The inspectors walked down the system to review as appropriate, mechanical and electrical equipment lineups, electrical power availability, system pressure and temperature indications, component labeling, component lubrication, component and equipment cooling, hangers and supports, operability of support systems, and to ensure that ancillary equipment or debris did not interfere with equipment operation. A review of a sample of past and outstanding work orders was performed to determine whether any deficiencies significantly affected the system function. In addition, the inspectors reviewed the 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 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 T-1b, Condensate Booster Pump Room – Elevation 712'0";
- Fire Zone CB-6, Main Control Room Complex – Elevation 800'0";
- Fire Zone A-3f, Division 2 Switchgear Room – Elevation 781'0";
- Fire Zone A-2n, Division 1 Switchgear Room – Elevation 781'0";
- Fire Zone CB-1c, General Access and HVAC [Heating, Ventilation, and Air Conditioning] Area – Elevation 719'0"; and
- Fire Zone D-5, Division 1 Diesel Generator Room – Elevation 737'0".

The inspectors verified that transient combustibles and ignition sources were appropriately controlled and assessed the material condition of fire suppression systems, manual 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

No findings were identified.

1R06 Flooding Protection Measures (71111.06)

.1 Internal Flooding

a. Inspection Scope

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

- Shutdown Service Water (SX) Pump Rooms; and
- Turbine Building Basement – Elevation 702'0".

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

b. Findings

(1) Apparent Nonconforming Condition Affecting Circulating Water (CW) Pump Auto Stop Feature during a Flooding Event

Introduction

The inspectors opened an Unresolved Item (URI) pending evaluation of an issue that affects the function of a flood protection design feature intended to automatically stop running CW pumps in the event of flooding in the Turbine Building Condenser Pit.

Discussion

As described in the UFSAR (Sections D3.6.4 and 10.4.5.5), each condenser cavity designed to contain flooding to Elevation 715' is equipped with a redundant system of level switches, which will alarm in the Control Room if the water level in the condenser cavity reaches an elevation of more than 1 foot (Elevation 710') above the condenser cavity floor at Elevation 709'. These level switches will close a motor-operated valve in the floor drain piping between the condenser cavity and the Turbine Building floor drain sump to slow flooding of the Turbine Building. Isolating the condenser pit from the Turbine Building floor drain sump slows early flooding of the Turbine Building basement. A second system of redundant level switches will automatically stop the CW pumps if the flood water reaches an elevation of 714' within the condenser cavity. An additional foot, from Elevation 714' to Elevation 715' remains to contain the water flow due to coast down of the CW pumps after they are initially shut off. Neither set of level switches are safety related; however, they are important internal flooding mitigation features as described in the UFSAR.

During review of Action Requests (ARs) 01192988 and 01197763, the inspectors noted that operators discussed a known design issue affecting the function of the CW pump auto stop feature. If CW Pump 'A' has power removed then all CW pump tripping protection is lost during a flooding event. With power removed from CW Pump 'A', CW Pumps 'B' and 'C' would not auto stop during a flooding event because the tripping power comes through the CW Pump 'A' tripping fuses. This appeared to the inspectors to be a nonconformance with the UFSAR description under the above specified circumstance. This design issue has apparently been known by operators for many years, yet no engineering evaluation was performed for the condition and no corrective action other than a procedure enhancement has been initiated by the licensee to address it. As stated in AR 01197763: "Another lesson learned was to plan ahead for off service CW pumps and that if possible, do not have 'A' CW Pump tagged out due to this removes the CW pump trip on high-high CW Pit level. The procedure for isolating a waterbox requires tagging out the off-service CW pump to preclude inadvertent start. Having this trip defeated while performing an evolution with the potential to cause CW Pit flooding is contrary to good industrial safety practices."

The inspectors reviewed CPS 3113.01, "Circulating Water (CW)," Revision 37e, and noted that Step 4.11 described the condition. "The High-High level (~714') in the Condenser Pit CW Pump Trip Circuitry is electrically powered by 1CW01PA DC [direct current] control power. If the 'A' pump is de-energized, then no CW pump tripping will be received from condenser pit flooding. The 710' floor drain sump floats which give Main Control Room alarm and 1TF013 valve closure would be the only indication of flooding."

The inspectors discussed this issue with the licensee and in response to the inspectors' questions; the licensee initiated AR 01355130 to evaluate the design concern. This issue is considered to be an Unresolved Item pending additional review of the licensee's evaluation by the inspectors (**URI 05000461/2012002-01, Evaluation of Apparent Nonconforming Condition Affecting Circulating Water Pump Auto Stop Feature during a Flooding Event**).

1R07 Heat Sink Performance (71111.07)

.1 Annual Heat Sink Performance (71111.07A)

a. Inspection Scope

The inspectors reviewed the licensee's thermal performance testing of the RHR 'B' heat exchanger. The inspectors assessed the conduct of the test and the testing results to verify that no deficiencies existed that would adversely impact the heat exchanger's ability to transfer heat to the SX system and to ensure that the licensee was adequately addressing problems that could affect the performance of the heat exchanger.
The inspectors observed portions of testing and reviewed documentation to verify that the acceptance criteria specified in procedure CPS 2700.20, "RHR A(B) Heat Exchanger, 1E12B001A(B) Thermal Performance Test Covered by NRC Generic Letter 89-13," Revision 4b, were satisfactorily met.

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

b. Findings

No findings 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 March 14, 2012. 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 were identified.

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

a. Inspection Scope

On February 4, 2012, the inspectors observed licensed operators in the Control Room perform a power reduction to attempt recovery of a control rod that would not couple during plant start up from the refueling outage in December 2011. This was an activity that required heightened awareness, additional detailed planning, and involved increased operational risk. The inspectors evaluated the following areas:

- Licensed operator performance;
- Crew's clarity and formality of communications;
- Ability to take timely actions in the conservative direction;
- Prioritization, interpretation, and verification of annunciators;
- Correct use and implementation of procedures;
- Control board manipulations;
- Oversight and direction from supervisors; and
- Ability to identify and implement appropriate TS actions and Emergency Plan actions and notifications as applicable.

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

b. Findings

No findings 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):

- Division 1 DG Failed To Start During Testing;
- Low Pressure Coolant Injection (LPCI) from RHR 'A' Check Valve 1E12-F041A Failed As-Found Leak Rate Measurement Test; and
- Outboard Main Steam Isolation Valve Above Seat Drain Valve 1B21-F067B Failed To Shut During Testing.

The inspectors assessed performance issues with respect to the reliability, availability, and condition monitoring of the structures, systems, and components (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) Unacceptable Preconditioning of LPCI from RHR ‘A’ Check Valve Prior to Leak Rate Test Measurement

Introduction

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.” The licensee failed to establish an adequate procedure to perform required leak rate testing for the LPCI Train ‘A’ discharge check valve. Specifically, the surveillance test procedure resulted in unacceptable preconditioning of the valve prior to a leak rate test measurement due to improper test sequencing. In addition, the licensee failed to correctly evaluate a failed leak rate test of the valve.

Discussion

During the refueling outage on December 1, 2011, the licensee performed as-found leak rate measurement testing of reactor coolant system (RCS) pressure isolation valve (PIV) 1E12-F041A, LPCI from RHR ‘A’ Check Valve, in accordance with CPS 9843.01, “ISI [Inservice Inspection] Category ‘A’ Valve Leak Rate Test,” Revision 35. This surveillance test procedure was performed to satisfy TS Surveillance Requirement (TSSR) 3.4.6.1, which required the licensee to verify the equivalent leakage of each RCS PIV is ≤ 0.5 gallon-per-minute (gpm) per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure ≥ 1000 pounds per square inch gauge (psig) and ≤ 1025 psig in accordance with the Inservice Testing (IST) Program. The licensee’s IST Program specified testing this valve once every 24-month refueling cycle during an outage. As described in the Bases for TS 3.4.6.1, the main purpose in establishing a leakage limit for the RCS PIVs is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication of whether the PIVs between the RCS and the connecting systems are degraded or degrading. 1E12-F041A is also a primary containment isolation valve and must satisfy TSSR 3.6.1.3.10, which required the licensee to verify the combined leakage rate through hydrostatically tested lines that penetrated the primary containment is within limits established in accordance with the Primary Containment Leakage Rate Testing Program stipulated by TS 5.5.13 and 10 CFR 50, Appendix J, “Primary Containment Leakage Testing for Light-Water Cooled Power Reactors.”

During the test, 1E12-F041A would not pressurize due to an excessive amount of water passing through the valve’s seat at low pressure. Since the valve would not pressurize, the leak rate was initially determined to exceed the TSSR 3.4.6.1 limit of 5 gpm and operators documented the test failure in AR 01296786. The licensee subsequently cycled the check valve open and then closed with a torque wrench on the valve’s mechanical exerciser linkage to fully seat the valve and satisfactorily retested it. The licensee completed an engineering evaluation (ECR 403032) following the initial leak rate test failure to review/accept the open stroke test as “maintenance/flushing” of the valve, which it deemed to satisfy the American Society of Mechanical Engineers (ASME) Operations and Maintenance (OM) Code requirements. The licensee also concluded in the engineering evaluation that the check valve was not preconditioned because “the valve closed by its own means and is not assisted into the seat by the torque wrench.”

The inspectors reviewed the licensee’s engineering evaluation for the failed as-found leak rate test and questioned whether the licensee correctly satisfied the ASME Code requirements. The inspectors noted that Paragraph ISTC-3630(f) of the ASME OM Code states, in part, that “Valves with leakage rates exceeding the values specified by the Owner shall be declared inoperable and either repaired or replaced. A retest demonstrating acceptable operation shall be performed following any required corrective action before the valve is returned to service.” It appeared to the inspectors that simply exercising the check valve to reseat it preconditioned the valve to pass a subsequent leak rate test, but it did not satisfy the Code requirement to “either repair or replace” the valve.

The inspectors discussed several questions regarding the testing of 1E12-F041A and the engineering evaluation with the licensee, which the licensee documented in AR 01326252:

1. Why did the check valve fail the as-found leak rate test? What was the cause?

2. How was the cause for the leak rate test failure corrected?

3. Did the licensee unacceptably precondition the valve by exercising it to get an acceptable leak rate test result?

4. How was cycling the check valve with a torque wrench considered a “repair” or “replacement” to satisfy the ASME Code requirement?

In response to the inspectors’ first two questions, the licensee determined that the check valve was unable to be pressurized for the leak rate test because the valve’s disc was not tightly seated due to previous testing performed on the redundant Train ‘A’ LPCI isolation valve (1E12-F042A, LPCI from RHR ‘A’ Shutoff Valve). 1E12-F042A is the upstream motor-operated isolation valve (MOV) on the containment penetration. The test on 1E12-F042A pressurized the volume between the MOV and the check valve, causing the check valve to slightly lift and remain off of its seat. Since the check valve was not tightly seated when it was tested, pressurizing the test fill volume during the initial as-found leak rate test could not provide flow equivalent to that seen during normal operation with RCS pressure against the check valve. As a result, the low pressure/volume test source water went through the unseated check valve rather than force the disc back tightly against the valve seat. Had the licensee performed the leak rate test on 1E12-F041A before testing 1E12-F042A, the check valve would not have been disturbed and should have passed the initial as-found leak rate test. CPS 9843.01 did not specify the testing sequence of the check valve and MOV (e.g., check valve before the MOV).

The inspectors noted that Inspection Manual Technical Guidance Part 9900 defines unacceptable preconditioning, in part, as: "The alteration, variation, manipulation, or adjustment of the physical condition of an SSC before or during TS surveillance or ASME Code testing that will alter one or more of an SSC's operational parameters, which results in acceptable test results. Such changes could mask the actual as found condition of the SSC and possibly result in an inability to verify the operability of the SSC. In addition, unacceptable preconditioning could make it difficult to determine whether the SSC would perform its intended function during an event in which the SSC might be needed." The Part 9900 Technical Guidance further states that influencing test outcome by performing valve stroking does not meet the intent of the as-found testing expectations described in NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," (April 1995), and may be unacceptable.

The inspectors also noted that manually cycling a check valve prior to performing an as-found leak rate measurement was also not in accordance with the licensee's Appendix J Program and IST Program procedural guidance. CPS 1305.01, "Primary Containment Leakage Rate Testing Program," Revision 10, Step 4.5, states, in part, "Preconditioning of components prior to leak rate testing is not allowed." In addition, ER-AA-321-1007, "IST Program Corporate Technical Positions," Attachment 1, "CTP-IST-001, Preconditioning of IST Program Components," Revision 0, states, in part, "Manipulation of a check valve or a vacuum breaker that uses a mechanical exerciser to measure breakaway force prior to surveillance testing would be unacceptable preconditioning."

In response to the inspectors' last two questions, the licensee noted that Paragraph ISTC-3630 states that valve closure before seat leakage testing shall be by using the valve operator with no additional closing force applied and concluded that exercising 1E12-F041A with a torque wrench on the mechanical exerciser linkage was acceptable preconditioning to establish the test conditions for an as-found leak rate test of the check valve since the previous test of 1E12-F042A had unseated the check valve; and, therefore its initial determination of an "as-found" test failure in AR 01296786 was incorrect. The licensee further concluded that exercising 1E12-F041A with a torque wrench was not a "repair" and would not meet the Code requirement; however, it was done to detect a possible valve problem that may require maintenance. The licensee therefore considered the initial leak rate measurement test to be an invalid test and the second leak rate measurement test to be the "as-found" test.

The inspectors acknowledged the licensee's position and agreed that the Code would allow valve closure before seat leakage testing; however, the inspectors believed that that provision is to allow for closing a normally open valve when necessary to establish conditions for testing it. In this case, had the licensee tested 1E12-F041A before testing 1E12-F042A, the check valve should have already been tightly seated and there would have been no need to exercise it closed. Therefore, the inspectors concluded that exercising check valve 1E12-F041A closed was unacceptable preconditioning prior to an as-found leak rate test measurement since it should have been unnecessary to do so and the licensee's surveillance test procedure was not appropriate to the circumstances because it did not properly sequence testing of 1E12-F041A before testing 1E12-F042A. Because exercising 1E12-F041A prior to measuring the leak rate unacceptably preconditioned the check valve, the as-found leak rate test results were invalid. The test results, however, could be considered an acceptable "as-left" leak rate measurement and; therefore, no safety concern remains for the next operating cycle.

In addition, the inspectors concluded that the initial engineering evaluation, ECR 403032, incorrectly evaluated the failed leak rate measurement test of 1E12-F041A. The engineering evaluation should have identified that the problem was due to the test sequencing and evaluated the preconditioning concern prior to exercising the check valve closed and re-performing the leak rate measurement. As stated in NUREG-1482, Section 3.5.2, "As noted in the inspection guidance, the licensee should have evaluated and documented the activity as acceptable preconditioning before performing the testing." Had it actually been a failed leak rate test rather than an invalid leak rate test, the inspectors also determined that ECR 403032 incorrectly concluded that exercising the check valve with a torque wrench on the mechanical exerciser met the Code requirement to "either repair or replace" the valve.

The licensee initiated several corrective actions to address this issue including:

1. Revision to CPS 9843.01 to ensure the correct testing sequence for RCS PIVs to test the check valve prior to testing the other inline valve to prevent lifting the check valve off its seat.

2. Revision to CPS 9843.01 to provide guidance for engineering evaluation of leak rate testing results that exceed the acceptance criteria to ensure the Corrective Action requirements in Paragraph ISTC-3630(f) to "either repair or replace" the valve are met.

3. Discussion with IST engineering peers regarding check valve exercising not satisfying the Code requirement for Corrective Action following a leak rate test failure.

4. Revision to applicable licensee IST procedure(s) to incorporate corrective actions to address the performance issues discussed above.

<u>Analysis</u>

The inspectors determined that the licensee's failure to establish an adequate surveillance test procedure to perform required leak rate testing for 1E12-F041A under suitable environmental conditions and to correctly evaluate the initial failed leak rate test 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 attribute for the containment 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, exercising the check valve prior to performing an as-found leak rate measurement masked the actual as-found condition of the valve, invalidating the test results. Because the preconditioning altered the as-found condition of the check valve, the data collected through the performance of the surveillance test was not fully indicative of the true valve performance trend. Additionally, the licensee's failure to correctly evaluate the initial failed leak rate test would become a more significant safety concern if left uncorrected because it could reasonably result in an unrecognized condition with a check valve failing to fulfill a safety-related function.

Therefore, this performance deficiency had a direct effect on the licensee's ability to fully assess the past operability, as well as the ability to trend as found data for the purpose of assessing the reliability of the check valve. The inspectors performed a Phase 1 SDP review of this finding using the guidance provided in IMC 0609, Attachment 0609.04, "Phase 1 – Initial Screening and Characterization of Findings." In accordance with Table 4a, "Characterization Worksheet for IE [Initiating Events], MS [Mitigating Systems], and BI [Barrier Integrity] Cornerstones," the inspectors determined that that this finding was a licensee performance deficiency of very low safety significance (Green) because the finding would not result in exceeding the TS limit for RCS leakage and would not have likely affected mitigation systems resulting in a loss of safety function. In addition, the finding did not represent an actual open pathway in the physical integrity of the reactor containment. Based on consultation and review with the Regional Senior Reactor Analyst, the inspectors concluded that the finding did not result in an increase in the likelihood of an initiating event such as an inter-system loss-of-coolant accident (LOCA) or a containment bypass event because the redundant isolation valve and closed loop system piping passed leak rate measurement testing during the refueling outage with considerable margin.

The inspectors concluded that this finding affected the cross-cutting area of human performance. Specifically, the licensee did not have adequately trained and knowledgeable personnel available to correctly evaluate the cause of the initial failed leak rate measurement test and to ensure that appropriate actions to correct the test sequence in the procedure were identified. (IMC 0310, H.2(b))

Enforcement

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

10 CFR 50, Appendix B, Criterion XI, "Test Control," requires, in part, that a test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met and that the test is performed under suitable environmental conditions.

Contrary to the above, surveillance test procedure CPS 9843.01, "ISI Category 'A' Valve Leak Rate Test," Revision 35, was not appropriate to the circumstances because it did not ensure leak rate measurement testing of LPCI from RHR 'A' Check Valve 1E12-F041A on December 1, 2011, was performed under suitable environmental conditions. Specifically, the test procedure did not appropriately ensure that the sequence of testing 1E12-F041A was prior to testing the redundant LPCI isolation valve (1E12-F042A, LPCI from RHR 'A' Shutoff Valve). This resulted in an invalid as-found leak rate measurement and unacceptable preconditioning of 1E12-F041A in order to re-perform the leak rate test measurement. Because of the very low safety significance, this violation is being treated as a non-cited violation consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000461/2012002-02, Unacceptable Preconditioning**

of Low Pressure Coolant Injection from Residual Heat Removal 'A' Check Valve Prior to Leak Rate Test Measurement). The licensee entered this violation into its corrective action program as AR 01326252 and AR 01355132.

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 18 on the Division 3 DG;
- Emergent maintenance during the week of January 23-27 on the 1A Reheater Drain Tank Level Controller, Control Room Ventilation Train 'B' Chiller, and on the 4160 Volt Bus 1A1 DG feeder breaker;
- Planned maintenance during the week of February 6-10 on the Division 2 DG, Spent Fuel Pool Cooling Pump 'B', Standby Gas Treatment Train 'B', Diving Operations at the Lake Screen House, and Co-60 Project Cask Loading and Transportation;
- Planned maintenance during the week of February 27 – March 2 on Containment Spray Train 'B', the Division 1 DG, Standby Gas Treatment Train 'B', and emergent maintenance on Control Room Ventilation Train 'A' Chiller and Division 1 DG Ventilation System; and
- Planned maintenance during the week of March 19-23 on the Division 3 DG and SX System.

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

b. Findings

No findings were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following issues:

- AR 01285559, “Incorrect Actuator Weight in Seismic Qualification of MOV";
- AR 01308448, “C1R13LL – No Activity To Place 1RIX–PR008A-D In Service”;
- AR 01304146, “Circulating Water Level Floats Partially Failed 3813.01 Section 8.1.3";
- EC 387458, “Impact of Transient Items in the Plant,” and EC 387241, “Transient Items Inside Primary Containment During Mode 3 (Prior to C1R13)”; and
- EC 387664, “AR 01323827: Potential Design Vulnerability: Single Open Phase.”

The inspectors selected these potential operability/functionality 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. When applicable, the inspectors also verified that the licensee appropriately assessed the functionality of SSCs that perform specified functions described in the UFSAR, Operations Requirements Manual, Emergency Plan, Fire Protection Plan, regulatory commitments, or other elements of the current licensing basis when degraded or nonconforming conditions were identified.

In addition, the inspectors verified that problems related to the operability or functionality 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 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:

- WO 01278457-01, “1SX012A Thrust Verification and MOV Clean/Inspect";
- WO 01278458-01, “1SX062A Thrust Verification and MOV Clean/Inspect";
- WO 01517195-03, "Control Room HVAC Chiller B";
- WO 01324596-01, "Replace / Bench Test / Adjust 1DO05A";
- WO 01318718-01, "C1R13 Valcor Valve Modifications 1PS038"; and
- WO 01347808-02, "Operations Post-Maintenance Test 0SA01D Dryer."

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

b. Findings

No findings were identified.

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 9000.01D001, “Control Room Surveillance Log – Mode 1,2,3 Data Sheet, “Section 8.9, “Reactor Coolant System – Operational Leakage”; (RCS Leakage)
- CPS 9070.01, “Control Room Heating Ventilation and Air Conditioning Air Filter Package Operability Test Run”; (Routine Test)
- CPS 9080.12, "Diesel Generator Fuel Oil Transfer Pump Operability"; (Inservice Test)
- CPS 9080.14, “Diesel Generator 1C 24-Hour Run and Hot Restart – Operability"; (Routine Test)
- CPS 9053.04C001, "RHR Loop A Valve Operability"; and (Inservice Test)
- CPS 9061.01, “Shutdown Service Water Operability Test.” (Inservice Test)

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 one RCS leakage test, three in-service tests, and two routine surveillance tests for a total of six surveillance testing inspection samples as defined in IP 71111.22.

b. Findings

No findings were identified.

2. RADIATION SAFETY

Cornerstone: Public Radiation Safety

2RS1 Radiological Hazard Assessment and Exposure Controls (71124.01)

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

.1 Radiological Hazard Assessment (02.02)

a. Inspection Scope

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

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

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.

- Cobalt-60 Harvesting and Packaging; and
- RHR Valve Refurbishment.

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 (This evaluation may include licensee planned entry into non-routinely entered areas subject to previous contamination from failed fuel.);
- The hazards associated with work activities that could suddenly and severely increase radiological conditions and that the licensee has 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.

b. Findings

No findings were identified.

.2 Instructions to Workers (02.03)

a. Inspection Scope

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

The inspectors reviewed the following radiation work permits used to access high radiation areas and evaluated the specified work control instructions or control barriers:

- RWP 10012088, “Cobalt-60 Harvesting"; and
- RWP 10013165, “2012 RHR / LPCS [Low Pressure Core Spray] / HPCS [High Pressure Core Spray] / SX / RCIC [Reactor Core Isolation Cooling] Scope of Work.”

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

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

For work activities that could suddenly and severely increase radiological conditions, the inspectors assessed the licensee’s means to inform workers of changes that could significantly impact their occupational dose.

b. Findings

Introduction

The inspectors identified an Unresolved Item concerning events that occurred on December 17, 2011, when three workers received electronic dosimeter (ED) dose rate alarms while working in the reactor cavity.

Discussion

The workers were lifting and setting the drywell head bolts in the reactor cavity, an area with high levels of radioactive contamination. The handling of these bolts caused contamination to build up on the workers' gloves. The contamination levels on the gloves became high enough and caused the ED worn on the chest to alarm. Two of the alarms occurred when the worker handled the ED to read the accumulated dose. The third worker alarm occurred when his hands came close to the chest and the ED reported dose rates of 1195 mrem/hour deep dose and 1405 mrem/hour shallow dose. The initial investigation performed by the licensee evaluated the whole body dose to the workers. This investigation determined that the workers were briefed to receive dose rate alarms, therefore the events were not entered into the licensee's corrective action program. At the time of the inspection, the licensee had not completed an evaluation of the radiological dose to the extremity (hands) from the build-up of contamination on the gloves worn by the workers that caused the ED alarms. Additionally, the licensee could not demonstrate the radiological controls that were in place when the alarms occurred. The issue is categorized as an Unresolved Item pending completion of a revised evaluation and the NRC's review of it (**URI 05000461/2012002-03, Incomplete ED Dose Rate Alarm Evaluation**).

.3 Contamination and Radioactive Material Control (02.04)

a. Inspection Scope

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

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

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

The inspectors selected several sealed sources from the licensee's inventory records and assessed whether the sources were accounted for and verified to be intact.

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

b. Findings

No findings were identified.

.4 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, radiation work permits, and worker briefings.

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

The inspectors examined the posting and physical controls for selected high radiation areas and very-high radiation areas to verify conformance with the occupational performance indicator.

b. Findings

No findings were identified.

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

a. Inspection Scope

The inspectors discussed with the Radiation Protection Manager the controls and procedures for high-risk high radiation areas and very-high radiation areas.
The inspectors discussed methods employed by the licensee to provide stricter control of very-high radiation area access as specified in 10 CFR 20.1602, "Control of Access to Very-High Radiation Areas," and Regulatory Guide 8.38, "Control of Access to High and Very-High Radiation Areas of Nuclear Plants." The inspectors assessed whether any changes to licensee procedures substantially reduce the effectiveness and level of worker protection.

The inspectors discussed the controls in place for special areas that have the potential to become very-high radiation areas during certain plant operations with first-line health physics supervisors (or equivalent positions having backshift health physics oversight authority). The inspectors assessed whether these plant operations require communication beforehand with the health physics group, so as to allow corresponding timely actions to properly post, control, and monitor the radiation hazards including re-access authorization.

The inspectors evaluated licensee controls for very-high radiation areas and areas with the potential to become radiation areas to ensure that an individual was not able to gain unauthorized access to the very-high radiation area.

b. Findings

No findings were identified.

.6 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 radiological conditions in their workplace and the radiation work permit controls/limits in place, and whether their performance reflected the level of radiological hazards present.

b. Findings

No findings were identified.

.7 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 and the radiation work permit 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 were identified.

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation (71124.08)

This inspection constituted a partial sample as defined in IP 71124.08.

.1 Shipment Preparation (02.05)

a. Inspection Scope

The inspectors observed shipment packaging, surveying, labeling, emergency instructions, shipping papers, and licensee verification of shipment readiness. The inspectors assessed whether the requirements of applicable transport cask certificate of compliance had been met. The inspectors evaluated whether the receiving licensee was authorized to receive the shipment packages. The inspectors evaluated whether the licensee's procedures for cask loading and closure procedures were consistent with the vendor's current approved procedures.

The inspectors observed radiation workers during the conduct of radioactive waste processing and radioactive material shipment preparation and receipt activities. The inspectors assessed whether the shippers were knowledgeable of the shipping regulations and whether shipping personnel demonstrated adequate skills to accomplish the package preparation requirements for public transport with respect to:

- The licensee's response to NRC Bulletin 79-19, "Packaging of Low-Level Radioactive Waste for Transport and Burial," dated August 10, 1979; and
- Title 49 CFR Part 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communication, Emergency Response Information, Training Requirements, and Security Plans," Subpart H, "Training."

b. Findings

No findings were identified.

4. OTHER ACTIVITIES

4OA1 Performance Indicator Verification (71151)

.1 Review of Submitted Quarterly Data

a. Inspection Scope

The inspectors performed a review of the data submitted by the licensee for the Fourth Quarter 2011 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 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 Licensee Event Report (LER) from

January 1, 2011, through December 31, 2011, 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. The inspectors noted that there was one unplanned scram in 2011.

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

b. Findings

No findings 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, 2011, through December 31, 2011, 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. The inspectors noted that there were no unplanned scrams with complications in 2011.

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

b. Findings

No findings were identified.

.4 Unplanned Power Changes per 7000 Critical Hours

a. Inspection Scope

The inspectors verified the Unplanned Power Changes per 7000 Critical Hours Performance Indicator for Unit 1. The inspectors reviewed power history data from January 1, 2011, through December 31, 2011, 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. The inspectors noted that there were no unplanned power changes in 2011.

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

b. Findings

No findings 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, 2011, through December 31, 2011, 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.

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

b. Findings

No findings were identified.

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 were identified.

.2 Semi-Annual Trend Review

a. Inspection Scope

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

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

b. Assessment and Observations

(1) Overall Effectiveness of Trending Program

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

(2) Adverse Trend in Maintaining Acceptable Environmental Conditions in the Records Storage Vault

The inspectors noted that the licensee's records management staff has documented numerous (at least 24) action requests over the past 1½ years identifying deviations from temperature and humidity limits for the records storage vault and questioned whether the licensee was adequately addressing this relatively long-standing condition adverse to quality. Maintaining appropriate environmental conditions in the records storage vault is important to prevent long-term degradation of quality records stored within it, and as such, is required by the licensee's procedures that implement 10 CFR 50, Appendix B, Criterion II, "Quality Assurance Program," including NO-AA-10, "Quality Assurance Topical Report," Revision 86; and, RM-AA-101-1008, "Processing and Storage of Records," Revision 5. These procedures reference ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities," 1994, which the licensee committed to comply with in UFSAR Chapter 17, "Quality Assurance." NQA-1, Supplement 17S-1, "Supplementary Requirements for Quality Assurance Records," Section 4.4, "Storage Facilities," states, in part, that records shall be stored in facilities constructed and maintained in a manner which minimizes the risk of damage or destruction from environmental conditions such as high and low temperatures and humidity.

Based on review of relevant action requests, material condition and work history of the records storage vault HVAC unit and humidifier, and discussions with several of the licensee's staff, the inspectors determined that the licensee has not been timely in correcting this problem. The inspectors reviewed the licensee's functionality assessment of this condition that was recently completed on February 24, 2012. While this functionality assessment was not completed in a very timely manner, it did appropriately evaluate the nonconforming condition. An earlier functionality assessment was also completed in October 2011 when humidity in the vault was high. That assessment concluded that the vault HVAC system was not controlling relative humidity consistent with UFSAR Section 9.4.12.1.2 values. Because there has been no discernable damage to date of quality records in the vault attributable to temperature and humidity the inspectors concluded that this issue was of minor significance. The licensee currently has scheduled replacement of the records storage vault HVAC unit in May 2012.

(3) <u>Continuing Adverse Trend in Evaluating Degraded/Nonconforming Plant Conditions for Operability, Functionality and/or Reportability</u>

The inspectors noted that an adverse trend has continued involving the licensee's evaluation of degraded/nonconforming plant conditions for operability, functionality and/or reportability. The inspectors have identified several examples of poor quality evaluations as well as the absence of evaluations for degraded/nonconforming conditions during this inspection period. The inspectors first identified and documented this adverse trend three years ago. Past semi-annual trend reviews documented in inspection reports (NRC Inspection Reports 05000461/2011004, 05000461/2010005 and 05000461/2009005) discussed examples of deficiencies with licensee evaluations when degraded or nonconforming conditions were discovered. The licensee's Nuclear Oversight organization has also noted this adverse performance trend in the past and has documented many specific examples of it. An Elevation Letter was issued by Nuclear Oversight in 2010 and was subsequently closed after corrective actions were implemented.

The inspectors have documented several findings related to this adverse performance trend over the past three years. These findings include FIN 05000461/2009003-01, NCV 05000461/2009004-03, NCV 05000461/2010003-01, FIN 05000461/2010003-03, NCV 05000461/2011004-04, FIN 05000461/2011004-05. Additional occurrences of incomplete or inadequate operability, functionality and/or reportability evaluations were identified by the inspectors, but were not documented because the individual issues were determined to be of minor safety significance.

Additional examples the inspectors identified during this inspection period, which were determined to be of minor safety significance included:

- AR 01300208 - No functionality assessment was performed as requested by the Operations Shift Manager in the action request for a degraded/non-conforming condition affecting operation of sump pumps in all three SX Pump Rooms.
- AR 01192988 – An apparent nonconforming condition involving the design of a flood protection feature for auto stopping the CW pumps has been known by plant operators for many years as identified in the action request. This issue is further discussed in Section 1R06.1 of this inspection report. No evaluation was performed for the condition and no corrective action other than a procedure enhancement was initiated by the licensee to address it.
- AR 01315336 and AR 01337692 – During a Diesel Fire Pump 'A' surveillance test; engine coolant temperature never reached the normal operating range of 160°F to 200°F. The acceptance criterion in Step 9.2.2.1 of the surveillance test procedure was not met, yet the surveillance test was signed off by operators as satisfactory with no explanation in the test record and no functionality assessment was performed.
- AR 01302633 and AR 01324027 – Motor-operated valve 1B21-F067B (Train 'B' Main Steam Line Drain Valve) failed to shut during surveillance testing. The inspectors reviewed the licensee's past operability/reportability evaluation and identified several quality issues with it, including incorrect assumptions and

information (e.g., the valve is normally opened vice closed and the cause of the problem was attributed to torque switch contacts vice stem grease).

- AR 01304146 – A functional test to verify the function of flood protection level switches in the Turbine Building Condenser Pit was completed at the end of the C1R13 refueling outage with an unsatisfactory result. The licensee deferred correcting the problem prior to start-up without performing an evaluation of the non-conforming condition and providing explicit justification as required by OP-AA-108-115, “Operability Determinations,” Revision 11. The functionality assessment completed by system engineering after the refueling outage did not correctly evaluate the condition; however, a condition evaluation completed by the Instrument Maintenance Manager provided an appropriate evaluation and explanation of the unsatisfactory test results.

Due to the fact that examples of this adverse performance trend continue to be identified by the inspectors as well as by the licensee and they have been entered into the licensee’s corrective action program, and that separate findings have been documented when an inadequate evaluation has risen to a more than minor significance threshold, no additional finding of significance was identified at this time.

.3 Annual In-depth Review Samples

a. Inspection Scope

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

- AR 01309488, “1TGCV1: ECR [Engineering Change Request] for TCV#1 [Turbine Control Valve #1] Oscillations and Impact to EH [Electro-Hydraulic] Pressure";
- AR 01297701, "Inadvertent Actuation Of Level 1 & 2 Logic"; and
- AR 01295617-05, "Automatic Scram on High Pressure During Approach to Unit Shutdown."

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

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

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

This inspection constituted three annual in-depth review inspection samples as defined in IP 71152.

b. Findings and Observations

(1) Influence of Schedule Pressure on Decision to Accept Out-Of-Specification Calibration Data for a Turbine Control Valve Resulted in a Challenge to Operators During Plant Start Up from the Refueling Outage

Introduction

The inspectors reviewed the licensee's apparent cause evaluation for an inaccurate engineering assessment of a nonconforming condition affecting performance of the main turbine control valves (TCVs) during plant startup from the C1R13 refueling outage in December 2011. Although No findings were identified from the self-revealed event, the inspectors noted that scheduling pressure influenced the licensee's decision making to accept the condition rather than correct it during the refueling outage. The inspectors' observations from review of this issue are discussed below.

Discussion

During the C1R13 refueling outage, instrument maintenance craftsmen calibrated the response of the TCVs. After completing the valve calibrations, the as-left results for TCV#1 were not within the limits specified in the procedure. The licensee evaluated and accepted the out-of-specification calibration data for TCV#1 rather than correcting the valve's response by recalibrating the valve's response curve. As stated in the licensee's apparent cause evaluation, recalibration to bring the response curve back into specification would have required additional instrument maintenance resources and additional outage scope, potentially affecting the outage duration. An Engineering Change Request (ECR 0004031170) documented acceptance of the out-of-specification calibration data for TCV#1 and provided guidance to operators. Subsequently, on December 25th during power ascension at about 80% reactor power, the TCVs started oscillating severely. This caused an excessive demand on the TCV electro-hydraulic control (EHC) system (i.e., beyond the capacity of a single EHC pump) causing system pressure to drop significantly. This also caused reactor power to oscillate as indicated on the average power range monitors. Operators manually started the standby EHC pump to restore pressure. As operators raised power above about 83%, TCV oscillations subsided and operators were able to shut off the standby EHC pump.

Following this event, the licensee wrote AR 01309488 to document the flawed engineering assessment of the non-conforming condition during the refueling outage and completed an apparent cause evaluation to determine why the engineering evaluation failed to identify the adverse impact on the EHC system that challenged operators during the power ascension. The licensee concluded that the apparent cause was that the engineering team assembled to evaluate the as-left calibration results narrowly focused on the acceptability of the calibration data, which limited their analysis to pressure control concerns. The team did not evaluate the aggregate impact on the EHC system but limited their scope to TCV#1 as a component. As a result, they did not identify the possible unintended consequences. The licensee identified several corrective actions including: (1) reperformance of the engineering evaluation to assess the TCV oscillations at about 80% power with recommendations for operators to mitigate operational impact, (2) read and sign briefing materials for engineering staff to address lessons learned, and (3) revision to plant operating procedures with instructions to run two EHC pumps when the plant is operated near 80% reactor power.

The inspectors identified one particular aspect of the engineering team's evaluation that was mentioned in the apparent cause evaluation, but was not identified by the licensee as a contributing cause. It was apparent to the inspectors that schedule pressure influenced the decision making process that led to the licensee's decision to accept the out-of-specification calibration data for TCV#1 vice recalibrating the response curve to correct it. The engineering team's thought process apparently focused on how to justify accepting the non-conforming condition in order to avoid expending time and limited instrument maintenance resources near the end of the refueling outage. The inspectors noted that this thought process led to a non-conservative decision with an unintended consequence. As stated in the apparent cause evaluation:

> "This technical call was primarily focused on the acceptability of current TCV#1 readings and not require [sic] any rework if we can live with the current readings. The result of the discussion was that the TCV#1 as-left curve was acceptable for unit operation even though not as good as desired. A recommendation was made to make an attempt to recalibrate *if time permits* [emphasis added] in the outage. The Outage Control Center was notified about the possibility of rework. The scope was unable to be completed during C1R13."

Until the problem can be corrected during an outage, the licensee has proceduralized an operator workaround to start the standby EHC pump when operating near 80% reactor power to avoid loss of EHC pressure due to excessive TCV oscillations. Inasmuch as operators were able to correctly respond to the TCV oscillations and resultant loss of EHC pressure in accordance with plant response procedures, the inspectors concluded that this issue was of minor safety significance.

(2) Failure to Incorporate Operating Experience into Preventive Maintenance (PM) Activities Associated with Steam Bypass System Circuit Cards

Introduction

A finding of very low safety significance with an associated non-cited violation of 10 CFR\ 50.65 was self-revealed when the failure of a Bypass Valve Demand (BVD) card caused Turbine Bypass Valve (BPV) #1 to suddenly close and isolate steam flow from the reactor to the main condenser, causing reactor pressure to exceed the automatic reactor scram setpoint. The card failure occurred, in part, due to the licensee's failure to perform preventive maintenance on the component throughout the history of plant operation. The automatic scram occurred on November 29, 2011, with the unit at 16.1% power while operators were shutting down the reactor to commence a refueling outage.

Discussion

On November 29, 2011, operators were reducing main generator load to 50 Megawatts in order to trip the main turbine to start refueling outage C1R13. BPV #1 had begun opening and was approximately 8 percent open when it ramped closed contrary to system demand. Due to the loss of the steam flow path through the bypass valve to the main condenser, reactor pressure rose to the automatic scram setpoint resulting in a reactor scram. Troubleshooting determined that BPV #1 closed due to the failure of a BVD card in the Steam Bypass and Pressure Control (SBPC) system. The cause of the card failure was attributed to the licensee's failure to replace or refurbish the BVD card prior to it failing. The installed BVD cards were original plant equipment and were therefore greater than 24 years old. The licensee performed a calculation which determined these components have a mean time between failures of 20 years. The licensee's root cause report identified this as an end-of-life failure of the BVD card.

The inspectors questioned why no PM had ever been performed on these circuit cards. The inspectors were told that the components were scoped within the Maintenance Rule, which requires that licensees monitor the performance of SSCs sufficient to provide reasonable assurance that these SSCs are capable of fulfilling their intended functions. These particular circuit cards had never been assigned equipment identification numbers and consequently had no classification within the licensee's PM process. Components of this type are not evaluated by the licensee for PM activities. By default, components such as these BVD cards are treated as run-to-failure and are only evaluated when their failure becomes classified as a Maintenance Rule Functional Failure. In such cases, the monitoring performed by the licensee in order to satisfy the requirements of 10 CFR 50.65 is performed at the plant level. This is permissible within the guidance provided in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2, which states, in part, under Section 9.3.2, "Remaining non-risk significant SSCs (those normally operating) are addressed under (a)(2) and performance is monitored against plant level criteria…. For example, automatic reactor scrams may be established as the performance criteria that are to be monitored to demonstrate the effectiveness of preventive maintenance for a given system."

However, Section 8.2.1.5 of NUMARC 93-01 also provides specific guidance for non-safety related SSCs whose failure causes a reactor scram such as the BVD card that failed on November 29th. It states, in part, that, "A utility should rely on actual plant-specific and industry wide operating experience… [This] operating experience is reviewed for plant-specific applicability and, where appropriate, is included in utility specific programs and procedures. It is appropriate to use this information to the extent practical to preclude unacceptable performance experienced in the industry from being repeated." Section 12.1 of this guidance also states that adjustment in PM activities shall be made under required 10 CFR 50.65(a)(3) reviews where necessary to ensure that the objective of preventing failures of SSCs through maintenance is appropriately balanced with minimizing unavailability. The licensee's root cause report for the November 29th scram identified two operating experience driven projects that it performed in 2001 and 2007. These Plant Material Condition Excellence Initiative (PMCEI) assessments failed to correctly classify these critical circuit cards within the licensee's PM process. In addition, AR 01117617, "Lack of PMs for Steam Bypass System Circuit Cards," documented a more recent failure by the licensee to incorporate this same operating experience into its PM activities. In the action request, the system engineer identified that troubleshooting BPV position transmitter cards revealed that the circuit cards had been replaced during the PMCEI Project, but no periodic PM was ever generated. He also noted that there were no PMs for periodic replacement of other circuit cards within the panel.

The recommended action of AR 011176 was for engineering to evaluate the circuit cards in the SBPC system for periodic replacement. The action was to be completed by November 18, 2010, but was delayed on three separate occasions due to "other emergent actions" and "other priorities." Eventually, on May 20, 2011, the issue was closed by an action to create a one-time replacement work order to be performed in

2014. The system engineer noted in the action request that the life of the circuit cards is approximately 16 years. At that time, the cards had been in service for 23 years.

The licensee initiated corrective actions to replace both BVD cards during the refueling outage, perform periodic replacement/refurbishment maintenance activities, and trend circuit card performance during routine calibration. Refer to Section 4OA3.3 of this inspection report for a review and closure of the LER associated with the reactor scram.

Analysis

The inspectors determined that the licensee's failure to evaluate and take into account, where practical, industry operating experience associated with preventive maintenance on critical component circuit cards in the SBPC system 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 this issue sufficiently similar to guidance provided in Example 7(c) in that this violation of 10 CFR 50.65(a)(3) had a consequence "…such as equipment problems attributable to failure to take industry operating experience into account when practicable." Specifically, industry operating experience driven reviews were performed focused upon critical circuit card replacement to prevent age-related failures and the licensee failed to apply this knowledge to the BVD cards on three separate occasions, eventually resulting in a reactor scram due to age-related failure. Therefore, the inspectors concluded that this finding was of more than minor safety significance. Because the performance issue caused a reactor scram, the inspectors determined that the 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 4a, "Characterization Worksheet for IE, MS, and BI Cornerstones," the inspectors determined 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 loss-of-coolant accident initiator, (2) did not contribute to both the likelihood of a reactor scram 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 concluded that this finding affected the cross-cutting area of human performance. Specifically, in the area of work control, the licensee did not appropriately coordinate work activities by incorporating actions to plan work activities to support long-term equipment reliability by scheduling maintenance as more preventive than reactive. (IMC 0310 H.3(b))

Enforcement

10 CFR 50.65(a)(3) states, in part, that performance and condition monitoring activities and associated goals and preventive maintenance activities shall be evaluated at least every refueling cycle provided the interval between evaluations does not exceed 24 months. The evaluations shall take into account, where practical, industry-wide operating experience. Adjustments shall be made where necessary to ensure that the objective of preventing failures of SSCs through maintenance is appropriately balanced against the objective of minimizing unavailability of SSCs due to monitoring or preventive maintenance.

Contrary to the above, on three occasions, in 2001 and 2007 and most recently on September 24, 2011, the licensee failed to incorporate operating experience when it was practical to do so. Because of the very low safety significance, this violation is being treated as a non-cited violation consistent with Section 2.3.2 of the NRC Enforcement Policy **(NCV 05000461/2012002-04, Failure to Incorporate Operating Experience into Preventive Maintenance Activities)**. The licensee entered this violation into its corrective action program as AR 01295617.

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

.1 (Closed) LER 05000461/2011-005-00, "Missed Surveillance Due To Preconditioning Valve Prior To Leak Rate Test"

On December 1, 2011, the inspectors observed the licensee's performance of surveillance testing that was accomplished in accordance with CPS 9861.05D010, "RCIC Turbine Exhaust Water Leak Rate Test Data Sheet (S-MC039k12)," Revision 24. This surveillance test procedure was performed to satisfy TSSR 3.6.1.3.10 for RCIC turbine exhaust check valve 1E51-F040, which required the licensee to verify the combined leakage rate through hydrostatically tested lines that penetrated the primary containment is within limits established in accordance with the Primary Containment Leakage Rate Testing Program stipulated by TS 5.5.13 and 10 CFR 50, Appendix J, "Primary Containment Leakage Testing for Light-Water Cooled Power Reactors." This Category A check valve was also tested to satisfy the IST Program requirements in TS 5.5.6 and 10 CFR 50.55a, Paragraph f, "Inservice testing requirements." The test was intended to be an "as-found" leak rate measurement prior to the performance of maintenance during the refueling outage.

During test preparation, the inspectors observed an operator cycle the check valve from closed to open and then back to close. Exercising the valve in this manner using the installed mechanical exerciser just prior to measuring the leak rate was directed by Step 8.2.2.2.a of the test procedure. This unacceptably preconditioned the valve prior to conducting the leak rate measurement since it masked the as-found condition and it was not necessary to place the system in the configuration for testing. Consequently, the surveillance test result was invalid. An acceptable leak rate measurement was performed after maintenance was completed; therefore, no safety concern remains for the next operating cycle. The licensee initiated a corrective action to revise CPS 9861.05D010 to ensure that 1E51-F040 is not exercised prior to as-found leak rate testing and completed an extent of condition evaluation to review other leak rate testing procedures for possible unacceptable preconditioning practices. No additional examples were found.

The surveillance test was determined to be a missed surveillance, which the licensee subsequently reported as an operation or condition prohibited by the plant's TS in accordance with 10 CFR 50.73(a)(2)(i)(B). The inspectors previously reviewed this issue and documented a non-cited violation of 10 CFR 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings" in NRC Inspection Report 05000461/2011005 for the licensee's failure to establish an adequate procedure to perform required leak rate testing for the RCIC turbine exhaust check valve. The inspectors determined that the information provided in LER 05000461/2011-005-00 did not raise any new issues or change the conclusion of the initial review. Therefore, the violation of TSSR 3.6.1.3.10 and TS 5.5.6 described in this LER will not be separately documented.

LER 05000461/2011-005-00 is closed.

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

.2 <u>(Closed) LER 05000461/2011-006-00, "Condition Prohibited by Technical Specifications Due to Missed Surveillance"</u>

During the refueling outage on December 7, 2011, the licensee performed leak rate measurement testing of RCS PIV 1E12-F042C, LPCI from RHR 'C' Shutoff Valve, in accordance with CPS 9843.01, "ISI Category 'A' Valve Leak Rate Test," Revision 35. This surveillance test procedure was performed to satisfy TSSR 3.4.6.1, which required the licensee to verify the equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure ≥ 1000 pounds per square inch gauge (psig) and ≤ 1025 psig in accordance with the IST Program. 1E12-F042C is also a primary containment isolation valve and must satisfy TSSR 3.6.1.3.10, which required the licensee to verify the combined leakage rate through hydrostatically tested lines that penetrated the primary containment is within limits established in accordance with the Primary Containment Leakage Rate Testing Program stipulated by TS 5.5.13 and 10 CFR 50, Appendix J, "Primary Containment Leakage Testing for Light-Water Cooled Power Reactors." During the test, the valve would not pressurize due to an excessive amount of water passing through the seat at low pressure. Since the valve would not pressurize, the leak rate was determined to exceed the TSSR 3.4.6.1 limit of 5 gpm. The licensee subsequently repaired the valve and satisfactorily retested it. The licensee determined the cause to be due to wear on the guide ribs and excessive disc to rib clearances, such that the disc became cocked when closing and it did not fully seat.

In April 2011, the inspectors reviewed CPS 9843.01 and the completed test packages for RCS PIV testing performed during the previous refueling outage in January/February 2010 and identified that the licensee had failed to correctly incorporate the required test pressure limits of the TSSR into the surveillance test procedure and subsequently tested multiple RCS PIVs, including 1E12-F042C, at pressures greater than the maximum test pressure of 1025 psig, invalidating the testing. The licensee correctly addressed these missed TS surveillances in accordance with TSSR 3.0.3. The inspectors reviewed this issue and documented a non-cited violation of TSSR 3.4.6.1 in NRC Inspection Report 05000461/2011003.

Because leak rate testing of 1E12-F042C during the December 2011 refueling outage failed to meet the TSSR 3.4.6.1 limit, the valve was considered to have been inoperable and TS 3.4.6, which required the RCS PIV leakage to be within the specified limit with Unit 1 operating in Modes 1 and 2, not met for the previous operating cycle. The licensee reported this as an operation or condition prohibited by the plant's TS in accordance with 10 CFR 50.73(a)(2)(i)(B). The inspectors determined that the information provided in LER 05000461/2011-006-00 did not raise any new issues or change the conclusion of the initial review. A licensee identified non-cited violation of TS 3.4.6 and TS 3.6.1.3 is discussed in Section 4OA7.1 of this inspection report.

LER 05000461/2011-006-00 is closed.

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

.3 <u>(Closed) LER 05000461/2011-004-00, "Automatic Reactor Scram During Removal of Main Generator"</u>

On November 29, 2011, operators were reducing main generator load to 50 Megawatts in order to trip the main turbine to start refueling outage C1R13. BPV #1 had begun opening and was approximately 8 percent open when it ramped closed contrary to system demand. Due to the loss of the steam flow path through the BPV to the main condenser, reactor pressure rose to the automatic scram setpoint of 1065 psig, resulting in a reactor scram. Operators shortly established control of reactor pressure by taking manual control of the BPV utilizing the BPV opening jack. No safety relief valves lifted as a result of the pressure increase. The licensee reported this event as a condition that resulted in the automatic actuation of the reactor protection system in accordance with 10 CFR 50.73(a)(2)(iv)(A).

Two anomalies occurred during the transient. The first was a trip of the motor driven feedwater pump and main turbine at +52 inches reactor vessel narrow range level, and second was the absence of a reverse power generator trip. Subsequent review did not identify any failed or degraded components within the feedwater level control system. The motor driven feedwater pump and main turbine trip had no safety consequence. The consequence of the failure of a reverse power generator trip to occur was that the main generator was motored for approximately 90 seconds before operators manually tripped main generator output circuit breakers 4506 and 4510. Two reverse power relays did not actuate as expected, most likely due to high megavolt amps reactive output power and low megawatt input while the generator was motoring. Motoring the main generator for 90 seconds had no safety impact.

Troubleshooting determined that BPV #1 closed due to a BVD circuit card failure. The cause of the card failure was attributed to the licensee's failure to replace or refurbish the BVD card prior to age-related failure. The installed BVD cards were original plant equipment and therefore greater than 24 years old. A calculation performed by the licensee determined these components have a mean time between failures of 20 years. The inspectors reviewed the licensee's root cause investigation of the scram. This investigation identified two operating experience driven projects performed in 2001 and 2007. These assessments failed to correctly classify these critical BVD circuit cards within the licensee's PM process. It is noteworthy that the licensee's root cause investigation failed to address the recent AR 01117617, which identified the lack of PMs for the SBPC circuit cards. The performance issues related to this event are discussed in Section 4OA2.3.b.2 of this inspection report.

LER 05000461/2011-005-00 is closed.

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

.4 Licensee Event Notification 47544, "H2 [Hydrogen] Leak from Main Generator and NOUE [Notice of Unusual Event] Declared"

On December 21, 2011, during preparations to synchronize the main generator to the grid, the automatic voltage regulator (AVR) was placed into service. Concurrent with the field flash, main generator hydrogen pressure began to significantly decrease. Generator hydrogen pressure had been stable at 49 psig and reached its lowest value of 27.2 psig during the licensee's troubleshooting efforts to restore pressure. Personnel were dispatched to identify the source of the hydrogen leak. At 5:52 p.m. Central Standard Time (CST) reports from personnel in the Turbine Building identified high hydrogen levels and the Turbine Building was evacuated. An NOUE was declared at 5:57 p.m. CST and State notification was completed at 6:09 p.m. CST. Turbine Building roof vents were opened to ventilate the Turbine Building and all elevations were verified to be clear of hydrogen. Access to the Turbine Building was restored and the NOUE was terminated at 6:38 p.m. CST. The NRC was notified of the declaration of the NOUE at 6:56 p.m. CST.

The cause of the hydrogen leak was determined to be that a portion of a hydrogen seal became mispositioned and allowed hydrogen to be lost through the seal as a result of shaft movement when the AVR was energized. Movement of the turbine generator shaft during AVR startup will occur due to adding current to the field flash circuit. Leakage resulting from this cause would not be detected in the Turbine Building since the hydrogen would then travel through its design path through the seal oil system and out the roof vent. The inspectors concluded that given the circumstances of the event, the operating shift made an appropriately conservative declaration and responded adequately. However, the State notification was made at 6:09 p.m. CST and the NRC was notified at 6:56 p.m. CST, a difference of 47 minutes. Also, the event notification made to the NRC of the Unusual Event declaration was 59 minutes after the NOUE was declared. Per 10 CFR 50.72 (a)(1)(i) and 10 CFR 50.72 (a)(3), the licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes. The inspectors concluded that the issue of timeliness of notification was of minor safety significance. The issue was entered into the licensee's corrective action program as AR 01307258.

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

.5 Licensee Event Notification 47587, "Invalid Actuation of General Containment Isolation Signals"

On January 12, 2012, the licensee notified the NRC via telephone that an invalid actuation of general containment isolation signals had occurred affecting containment isolation valves in more than one system and emergency service water systems that normally do not run and that serve as ultimate heat sinks. The event occurred on December 4, 2011 at 6:09 a.m. CST during refueling outage activities. The telephone notification was made in accordance with 10 CFR 50.73(a)(1) and was within the 60-day reporting requirement.

The inspectors reviewed the event notification, prompt investigation, apparent cause evaluation (AR 01297701), and the action request documented for the event.

The inspectors noted several performance deficiencies associated with the event in the area of work control. Specifically, maintenance technicians exceeded the scope of work that was authorized, inadequate work instructions were accepted, and work activities proceeded on even after unexpected annunciator alarms caused by the event were received. However, no performance deficiency of more than minor safety significance was identified with this issue. This was primarily due to the fact that the plant was in a shutdown condition at the time of the event and the affected SSCs were not required to be operable.

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

.6 <u>(Closed) LER 05000461/2011-001-00, "Postulated Spurious High Pressure Core Spray Initiation Result Unanalyzed"</u>

On February 8, 2011, the licensee identified that the HPCS system could spuriously initiate due to fire-induced hot short cable damage to two automatic initiation logic instrument cables routed in the same raceway. As a result, the reactor would fill and flood the main steam lines because shorting of the control cables would prevent HPCS pump shutdown and closure of the injection valve. The licensee determined that the main steam relief valves and associated down-comers were not analyzed for stresses as a result of two-phase flow at high pressure.

Subsequent to the licensee's issuance of LER 05000461/2011-001-00, the licensee completed an analysis that determined the acceptability of a spurious HPCS initiation. The inspectors reviewed the analysis and did not identify any safety significant concerns. The concern identified in this LER was similar to the concern identified by the NRC in URI 05000461/2005006-01. Additional information regarding the analysis is discussed as part of the closure of URI 05000461/2005006-01 in Section 4OA5.3.

LER 05000461/2011-001-00 is closed.

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

4OA5 <u>Other Activities</u>

.1 <u>Review of Institute of Nuclear Power Operations (INPO) / World Association of Nuclear Operators (WANO) Assessment Report</u>

The inspectors completed a review of the INPO/WANO Evaluation Report for the Clinton Power Station assessment conducted in December 2010. During this review, the inspectors did not identify any new safety significant issues.

.2 <u>Review of INPO Training Accreditation Reports</u>

The inspectors completed a review of the INPO Operations Training Accreditation Board Report dated March 22, 2012, and the INPO Maintenance and Technical Training Accreditation Board Report dated June 10, 2009. During this review, the inspectors did not identify any new safety significant issues.

.3 (Closed) URI 05000461/2005006-01, "Postulated Fire-Induced Circuit Failures Resulting in Potential Spurious Actuation of Division III HPCS Pump 1E22-C001 and Discharge Valve 1E22-F004"

During the NRC's 2005 triennial fire protection baseline inspection the inspectors identified an URI associated with potential fire-induced electrical failures in the HPCS system. The inspectors postulated that a fire in the Division III switchgear room (located in Fire Zone CB-5a) could result in fire-induced electrical circuit faults in the control cables and control logic of the HPCS pump and a discharge valve. This could result in the inability to shut-off the pump and stop it from continuously injecting into the core. The inspectors requested that the licensee evaluate the potential scenario of a HPCS reactor overfill event and determine if the plant can achieve and maintain safe shutdown under such a scenario.

As a result of the inspectors' concerns, the licensee performed an evaluation as part of EC 386645, "Analysis of Reactor Overfill Impact on Main Steam Piping and SRVs [Safety Relief Valves] Due to Fire Induced Spurious HPCS Operation," Revision 0, and determined that safe shutdown would not be impacted under a HPCS reactor overfill event. The spurious operation of HPCS and a failure of discharge valve 1E22-F004 to close would result in continuous water flow into the reactor. Water and steam would flow through the SRVs rather than steam only as a result of a reactor overfill event.
The licensee determined in calculations IP-M-0792, "MSRV [Main Steam Relief Valve] Discharge Transient Force Analysis for Reactor Overfill," Revision 0, and IP-M-0793, "Piping Stress Analysis for SRV Discharge Transient Forces," Revision 0, that there would be no adverse impact on the main steam piping and SRVs as a result of a reactor overfill event.

The licensee concluded that no manual actions were required to control HPCS injection and that safe shutdown could be achieved during a HPCS reactor overfill event.
In calculation IP-M-0792, the licensee calculated the transient hydraulic forces generated in the main steam SRV discharge lines as a result of a reactor overfill event. The licensee then determined the impact of the transient hydraulic forces on the main steam piping in calculation IP-M-0793. The licensee determined that the SRVs and main steam piping would withstand stresses associated with increased flow of water resulting from a reactor overfill condition. The inspectors reviewed the analyses and the calculated stresses on the SRVs and main stream piping as a result of a HPCS reactor overfill event and did not identify any safety significant findings.

URI 05000461/2005006-01 is closed.

4OA6 Management Meetings

.1 Resident Inspectors' Exit Meeting

The inspectors presented the inspection results to W. Noll and other members of the licensee's staff at the conclusion of the inspection on April 19, 2012. The licensee acknowledged the findings presented. Proprietary information was examined during this inspection, but is not specifically discussed in this report.

.2 Interim Exit Meetings

Interim exit meetings were conducted for:

- The Radiological Hazard Assessment and Exposure Controls; Radioactive Solid Waste Processing; and Radioactive Material Handling, Storage, and Transportation Inspection with Mr. W. Noll and other members of the licensee's staff at the conclusion of the inspection on March 2, 2012.

- The review associated with URI 05000461/2005006-01, "Postulated Fire-Induced Circuit Failures Resulting in Potential Spurious Actuation of Division III HPCS Pump 1E22-C001 and Discharge Valve 1E22-F004," and LER 05000461/2011-001-00, "Postulated Spurious High Pressure Core Spray Initiation Result Unanalyzed," with Ms. K. Baker and other members of the licensee's staff on February 17, 2012.

4OA7 Licensee-Identified Violations

The following violation of very low safety significance (Green) was identified by the licensee and is a violation of NRC requirements, which meets the criteria of Section 2.3.2 of the NRC Enforcement Policy, NUREG-1600, for being dispositioned as a non-cited violation.

.1 Violation of TS 3.4.6 and TS 3.6.1.3 Due to Missed Surveillance of an RCS PIV

TS 3.4.6 requires, in part, that the leakage from each RCS PIV shall be within limits in Modes 1, 2, and 3. TS 3.4.6, Condition A, states that with one or more flow paths with leakage from one or more RCS PIVs not within the limit, isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve within 4 hours. TS 3.4.6, Condition B, states that if the required action and associated completion time of Condition A is not met, be in Mode 3 within 12 hours and Mode 4 within 36 hours.

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

Contrary to the above, RCS PIV 1E12-F042C was found with leakage in excess of the limit during leak rate measurement surveillance testing on December 7, 2011.
The licensee determined the cause for the failure was due to wear on the valve guide ribs and excessive disc to rib clearances, such that the disc became cocked when closing and it did not fully seat. In addition, testing of this valve during the previous operating cycle to satisfy TSSR 3.4.6.1 was not correctly performed, for which the licensee had taken action in accordance with TSSR 3.0.3. Based on the cause determination and the previous missed surveillance of 1E12-F042C, the inspectors concluded that this valve had not been capable of performing its specified safety function (and thus was inoperable) for a period of time before its discovery longer than allowed by TS 3.4.6 and TS 3.6.1.3. 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 would not result in exceeding the TS limit for RCS leakage and would not have likely affected mitigation systems resulting in a loss of safety function. In addition, the finding did not represent an actual open pathway in the physical integrity of the reactor containment. Based on consultation and review with the Regional Senior Reactor Analyst, the inspectors concluded that the valve test failure did not result in an increase in the likelihood of an initiating event such as an inter-system LOCA or a containment bypass event because the redundant isolation valve and closed loop system piping passed leak rate measurement testing during the refueling outage with considerable margin.

This violation of TS 3.4.6 and TS 3.6.1.3 is being treated as a non-cited violation consistent with Section 2.3.2 of the NRC Enforcement Policy. The licensee entered this violation into its corrective action program as AR 01305725. The licensee submitted LER 05000461/2011-006-00 to report this issue as a condition prohibited by TS. Refer to Section 4OA3.2 of this inspection report for the review and closure of the LER.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

R. Bair, Shift Operations Superintendent
K. Baker, Senior Manager Design Engineering
T. Chalmers, Work Management Director
J. Cunningham, Operations Director
C. Dunn, Regulatory Assurance Manager
R. Frantz, Regulatory Assurance
M. Heger, Mechanical Design Branch Manager
N. Hightower, Radiation Protection Operations Manager
D. Kemper, Engineering Director
K. Leffel, Operations Support Manager
S. Mohundro, Engineering Programs Manager
W. Noll, Site Vice President
T. Parrent, Fire Protection & Appendix J Program Engineer
J. Peterson, Regulatory Assurance
C. Rocha, Nuclear Oversight Manager
J. Smith, Senior Manager Plant Engineering
T. Stoner, Maintenance Director
J. Stovall, Radiation Protection Manager
B. Taber, Plant Manager
J. Ufert, Fire Marshall
T. Veitch, Chemistry Manager

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened

05000461/2012002-01	URI	Evaluation of Apparent Nonconforming Condition Affecting Circulating Water Pump Auto Stop Feature during a Flooding Event (Section 1R06.b.1)
05000461/2012002-02	NCV	Unacceptable Preconditioning of Low Pressure Coolant Injection from Residual Heat Removal 'A' Check Valve Prior to Leak Rate Test Measurement (Section 1R12.b.1)
05000461/2012002-03	URI	Incomplete ED Dose Rate Alarm Evaluation (Section 2RS1.2)
05000461/2012002-04	NCV	Failure to Incorporate Operating Experience into Preventive Maintenance Activities (Section 4OA2.3.b.2)

Closed

05000461/2012002-02	NCV	Unacceptable Preconditioning of Low Pressure Coolant Injection from Residual Heat Removal 'A' Check Valve Prior to Leak Rate Test Measurement (Section 1R12.b.1)
05000461/2012002-04	NCV	Failure to Incorporate Operating Experience into Preventive Maintenance Activities (Section 4OA2.3.b.2)
05000461/2011-005-00	LER	Missed Surveillance Due To Preconditioning Valve Prior To Leak Rate Test (Section 4OA3.1)
05000461/2011-006-00	LER	Condition Prohibited by Technical Specifications Due to Missed Surveillance (Section 4OA3.2)
05000461/2011-004-00	LER	Automatic Reactor Scram During Removal of Main Generator (Section 4OA3.3)
05000461/2011-01-00	LER	Postulated Spurious High Pressure Core Spray Initiation Result Unanalyzed (Section 4OA3.6)
05000461/2005006-01	URI	Postulated Fire-Induced Circuit Failures Resulting in Potential Spurious Actuation of Division III HPCS Pump 1E22-C001 and Discharge Valve 1E22-F004 (Section 4OA5.3)

Discussed

05000461/2009003-01	FIN	Failure to Evaluate Safety Function of Suppression Pool Makeup System (Section 4OA2.2.b.3)
05000461/2009004-03	NCV	Failure to Update the Final Safety Analysis Report (Section 4OA2.2.b.3)
05000461/2010003-01	NCV	Failure to Satisfy 10 CFR 50.72 and 50.73 Reporting Requirements (Section 4OA2.2.b.3)
05000461/2010003-03	FIN	Operability Assessment of Inservice Testing Surveillance Discrepancies for Excess Flow Check Valve (Section 4OA2.2.b.3)
05000461/2011004-04	NCV	Failure to meet Technical Specification 3.7.3 for Operability of Control Room Ventilation System (Section 4OA2.2.b.3)
05000461/2011004-05	FIN	Failure to Evaluate Operability of Control Room Ventilation System for Degraded Flow Condition (Section 4OA2.2.b.3)

05000461/2011005-04	NCV	Unacceptable Preconditioning of Reactor Core Isolation Cooling System Check Valve Prior to Leak Rate Test Measurement (Section 4OA3.1)
05000461/2011003-02	NCV	Failure to Meet Surveillance Testing Requirement for Reactor Coolant System Pressure Isolation Valves (Section 4OA3.2)

LIST OF DOCUMENTS REVIEWED

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

<u>1R04 Equipment Alignment</u>

- OP-AA-108-117, “Protected Equipment Program,” Revision 2
- CPS 3312.01E001, “Residual Heat Removal Electrical Lineup,” Revision 14
- CPS 3312.01, “Residual Heat Removal (RHR),” Revision 39e
- CPS 3312.01.V001, “Residual Heat Removal Valve Lineup,” Revision 17a
- CPS 3312.01V002, “Residual Heat Removal Instrument Valve Lineup,” Revision 9a
- CPS 3314.01, “Standby Liquid Control,” Revision 11c
- CPS 3314.01V001, “Standby Liquid Control Valve Lineup,” Revision 10
- CPS 3314.01V002, “Standby Liquid Control Instrument Valve Lineup,” Revision 6
- CPS 3314.01E001, “Standby Liquid Control Electrical Lineup,” Revision 9a
- CPS 3506.01C001, “Diesel Generator 1A Pre-Start Checklist,” Revision 14a
- CPS 3506.01P001, “Division 1 Diesel Generator Operations,” Revision 3b
- CPS 3506.01E001, “Diesel Generator and Support Systems Electrical Lineup,” Revision 18a
- CPS 3506.01V001, “Diesel Generator and Support Systems Valve Lineup,” Revision 13a
- CPS 3506.01V002, “Diesel Generator and Support Systems Instrument Valve Lineup,” Revision 11b
- CPS 9015.01, “Standby Liquid Control System Operability,” Revision 41
- CPS 9015.02, “Standby Liquid Control Injection Operability,” Revision 38
- CPS 9015.03, “Standby Liquid Control, Scram Discharge Volume Monthly Valve Verification,” Revision 25c
- ECR 402116, “SLC Suction Piping Pressure Verification,” Revision 0
- M05-1036, “P&ID Diesel Generator Fuel Oil System (DO),” Sheet 1, Revision S
- M05-1035, “Diesel Gen Aux System (DG) Starting Air Exhaust & combustion Sys,” Sheet 1, Revision AE
- M05-1075, “Residual Heat Removal, Sheet 1,” Revision AW
- M05-1075, “Residual Heat Removal, Sheet 2,” Revision AM
- M05-1075, “Residual Heat Removal; Sheet 4,” Revision AF
- M05-1077, “Standby Liquid Control (SC),” Sheet 1, Revision AB
- AR 01271103, “SLC Suction Pressure Gages Read Different”
- AR 01293160, “SLC Surveillance Change Enhancement – NRC Interface”
- AR 01312043, “Air was Observed During 9052.04 (RHR to Fuel Pool),”
- AR 01314711, 1E12F005: “Valve Failed Set Pressure,”
- AR 01321943, “NRC ID Protected Equipment Postings”
- AR 01323352, “Air Void Found on Line 1RH117A,”
- AR 01338731, “1E12F068A Stem Nut Wear Trending Up,”

<u>1R05 Fire Protection</u>

- Clinton Power Station Updated Final Safety Analysis Report, Appendix E, “Fire Protection Evaluation Report – Clinton Power Station Unit 1,” Revision 14
- CC-AA-211, “Fire Protection Program,” Revision 4
- OP-AA-201-009, “Control of Transient Combustible Material,” Revision 11

- OP-MW-201-007, “Fire Protection System Impairment Control,” Revision 7
- Calculation IP-M-0177, “Fire Loads in Clinton Power Station”
- CPS 1019.05, “Transient Equipment/Materials,” Revision 18c
- CPS 1893.04M003, “Clinton Power Station Pre-Fire Plan Legend,” Revision 1
- CPS 1893.04M132, “781 Auxiliary (East): Div 1 Switchgear Prefire Plan,” Revision 5
- CPS 1893.04M310, “719 Control: HVAC Equipment Area Prefire Plan,” Revision 6a
- CPS 1893.04M511, “737 Diesel Generator: Division 1 Diesel Generator and Day Tank Room Prefire Plan,” Revision 6a
- CPS 1893.04M701, “712 Turbine Condensate Booster Pump Room Prefire Plan,” Revision 6
- CPS 1893.04M362, “800 Control: MCR Support Offices & Corridor Prefire Plan,” Revision 5
- CPS 1893.04M361, “800 Control; Operations Support Center Prefire Plan,” Revision 5
- CPS 1893.04M360, “800 Control: Tech Support Center Prefire Plan, “Revision 5
- CPS 1893.04M130, “781-790 Auxiliary: Switchgear Prefire Plan,” Revision 5
- CPS 1893.04M364, “800 Control: Main Control Prefire Plan,” Revision 3
- AR 00987424, “Revise PFP 1893.04M310 to Add Sprinkler System 1FP39SA”

<u>1R06 Flooding Protection Measures</u>

- CPS Individual Plant Examination (IPE), Section 3.3.8, “Internal Flood Analysis,” September 1992
- CPS-PSA-012, “Clinton PRA 2003 Update Internal Flooding Update: Integration of the Internal Flooding Analysis into the Single-Top Model,” Revision 0
- Clinton Power Station Updated Safety Analysis Report, Revision 14
- 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
- A22-1032, “Circulating Water Screen House Main Floor Plan Area-12 – El. 699’0”,” Revision K
- CPS 3113.01, “Circulating Water (CW),” Revision 37e
- AR 01300208, “SX Pump Room Sumps Fail to Pump Down”
- AR 01192988, “Procedural Enhancement to CW 3113.01 Section 8.2.2”
- AR 01197763, “Lessons Learned from Downpower for Condenser Tube Leak”
- AR 01246805, “No Funding to Implement EC 382144”
- AR 01258570, “1LSCM282: EC 380120 Causes Spurious Flood Alarm 5065-5G”
- AR 01092206, “Functionality Review of Condenser Pit Level Switch”
- AR 01023891, “1LSTF001B Failed to Actuate Per 3813.01”
- AR 01304146, “CW Level Floats Partially Failed 3813.01 Section 8.1.3”
- AR 01355130, “NRC URI for CW Pump Trip Logic”

<u>1R07 Heat Sink Performance (71111.07)</u>

- NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” July 18, 1989
- Illinois Power Letter U-603130 Dated February 4, 1999, “Update to Illinois Power’s (IP’s) Response to Generic Letter 89-13”
- CPS 1003.10, “Clinton Power Station (CPS) Program for NRC Generic Letter 89-13 (Service Water System Problems Affecting Safety-Related Equipment),” Revision 6a
- CPS 2700.20, “RHR A(B) Heat Exchanger, 1E12B001A(B) Thermal Performance Test Covered By NRC Generic Letter 89-13,” Revision 4b
- WO 01244266-04, “OP Place RHR In Suppression Pool Cooling Mode” March 16, 2012
- AR 01340916, “ HOT Taps For RHR B/C Suction Cross Connect Piping”

1R11 Licensed Operator Requalification Program and Licensed Operator Performance

- OP-AA-101-113-1006, "4.0 Crew Critique Guidelines," Revision 3
- CL-2012-S-006, "Operations Actions to Couple Control Rod 28-33," Revision 0
- WO 1500914-02, "Operations Actions to Couple Control Rod 28-33," February 4, 2012
- AR 01322848, "Control Rod 28-33 Remains Uncoupled After WO 1500914-02"
- AR 01322894, "4.0 Crew D Critique on Re-coupling Attempt for 28-33"
- Control Room Logs, February 4, 2012

1R12 Maintenance Effectiveness

- Clinton Power Station Updated Safety Analysis Report, Revision 14
- Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2 March 1997
- NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2
- ER-AA-310, "Implementation of Maintenance Rule," Revision 8
- ER-AA-310-1001, "Maintenance Rule Scoping," Revision 4
- ER-AA-321-1007, "Inservice Testing (IST) Program Corporate Technical Positions," Revision 0
- EC 0000403032, "Maintenance In Support of Failed As-Found IST Close Test on 1E12-F041A (IR 01296786)"
- Operational and Technical Decision Making document 1176032, "1B21-F019, Main Steam Line Outboard Drain Isolation Valve," December 8, 2010
- CPS 3307.01, "Oscillation Power Range Monitor (OPRM)," Revision 0a
- CPS 9053.05, "RHR/LPCS Valve Operability (Shutdown)," Revision 41
- CPS 9061.08, "Main Steam System Drain Valves Operability," Revision 30
- CPS 9061.08D001, "Main Steam System Drain Valves Operability Data Sheet," Revision 30e
- CPS 9843.01, "ISI Category 'A' Valve Leak Rate Test," Revision 35c
- IST-CPS-BDOC-V-16, "CPS Inservice Testing (IST) Bases Document, Second Interval," Revision date 12/30/05
- IST-CPS-BDOC-V-23, "Clinton Inservice Testing Program Bases Document – Residual Heat Removal," June 10, 2010
- Illinois Power Letter U-601736 Dated September 27, 1990, "Illinois Power Company CPS Pump and Valve Testing Program Plan," Revision 8
- Exelon Nuclear Letter U-603967 Dated July 12, 2010, "Submittal of Inservice Testing Program Plan for the Third Ten-Year Interval," Revision date 6/28/2010
- CPS IST Program Plan Second Interval Cold Shutdown Justification CSJ-122, Revision 1
- WO 01350657-01, "Perform 9843.01 Category A Valve Leak Rate Test (1E12-F041A) LPCI A Drywell Isolation"
- WO 01350658-01, "Perform 9843.01 Category A Valve Leak Rate Test (1E12-F041A) LPCI A Drywell Isolation" (Retest)
- WO 01350666-01, "Perform 9053.05 RHR Valve Operability Shutdown (Section 8.6 for 1E12-F041A)"
- WO 01498241-01, "Troubleshoot to Determine Cause for MOV 1B21-F067B Failure to go Fully Closed Electrically," December 15, 2011
- Past Operability Evaluation 1302633-03, "1B21-F067B C1R12-C1R13," January 5, 2012
- M05-1002, "Main Steam (MS)," Sheet 1, Revision U
- M05-1002, "Main Steam (MS)," Sheet 2, Revision R
- M05-1070, "MSIV Leakage Control System (IS)," Revision AA
- AR 01297509, "1DG01KA: Division 1 DG Tripped Following Start"
- AR 01299156, "1DG01KB: Division 2 EDG Speed Sensor Amphanol [sic] Connector Loose"

- AR 01297512, “1DG01KA16: Division 1 DG Failed to Start”
- AR 01302633, “1B21-F067B Failed to Shut During 9061.08”
- AR 01323305, “1B21F067B: Lessons Learned for C1R13 – Document As-Found Condition”
- AR 01324009, “NRC Issues on 2nd IST Interval Cold Shutdown Justification”
- AR 01324027, “1B21-F067B NRC Interface – Related to IR 1302633”
- AR 01326252, “NRC Questions Regarding As-Found IST Leak Rate Test for 1E12-F041A”
- AR 01296786, “1E12-F041A Failed As Found Leak Rate”
- AR 01343842, “NRC Requests Additional Information From Work Group Evaluation for IR 134009”
- AR 01346128, “Benchmarking for IR 01313983”
- AR 01355132, “NRC NCV for Unnecessary Precondition of 1E12F041A During LRT”

1R13 Maintenance Risk Assessments and Emergent Work Control

- ER-AA-600, “Risk Management,” Revision 6
- ER-AA-600-1012, “Risk Management Documentation,” Revision 9
- ER-AA-600-1042, “On-Line Risk Management,” Revision 7
- WC-AA-101, “On-Line Work Control Process,” Revision 18
- WC-AA-104, “Integrated Risk Management,” Revision 18
- Clinton Power Station Technical Specifications
- CL-A4-09, “CPS Risk Analysis to Support Division 3 SX Header Unavailability with Reduced Online Risk Color,” Revision 0
- WO 01501634-01-01, “Reheater Drain Tank 1B Normal Drain Valve,” January 26, 2012
- AR 01306461, “Reheater Drain Valve 1HD012B Not Fully Closed Annunciator”
- AR 01316221, “1E32N653N: Annunciator 5067-4H Inboard MSIV LCS Inoperable”
- AR 01316543, “OPRM Module 1C51-N020B Inoperable and Reg B Card Power Fault”
- AR 01317600, “1LIC-HD030 RHDT 1A Lvl Controller Failing”
- AR 01318406, “Received Open Reading During 9333.20 Section 8.20”
- AR 01318419, “0VC13CB Motor Cooler Filter Housing Leaking Liquid Freon”
- AR 01332745, “Conflict Between Maintenance Rule & Paragon Delays Division 1 Battery Charger System Outage Window”

1R15 Operability Evaluations

- Clinton Power Station Technical Specifications
- Clinton Power Station Updated Final Safety Analysis Report, Revision 14
- LER 2010-004-00, “OPDRV Requirements Not Met During Control Rod Drive Mechanism Replacements”
- 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
- OP-AA-108-115, “Operability Determinations,” Revision 10
- EC 387664, “AR 01323827: Potential Design Vulnerability: Single Open Phase,” Revision 0
- EC 386361, “Analysis No. 19-AK-13, Analysis of Load Flow, Short Circuit, and Motor Starting Using ETAP Powerstation,” Revision 0
- EC 387241, “Transient Items Inside Primary Containment During Mode 3 (Prior to C1R13),” Revision 0
- EC 387458, “Impact Of Transient Items In The Plant (AR 1307329),” Revision 0
- EC 387501, “IR 01312043, Air Was Observed During 9052.04 (RHR A to Fuel Pool),” Revision 0
- CPS 1019.05, “Transient Equipment/Materials,” Revision 18b

- CPS 3007.01C005, “Operations with a Potential for Draining the Reactor Vessel Checklist,” Revision 2a
- CPS 9000.01D002, “Control Room Surveillance Log – Mode 4, 5,” Revision 37a, dates January 11 – January 17, 2010, January 18 – January 24, 2010
- AR 01308448, “C1R13 Lessons Learned – No Activity to Place 1RIX-PR008A-D In Service”
- AR 01316112, “Flexible Material In Containment”
- AR 01317250, “Review Past Operability of 1RIX-PR008A-D”
- AR 01322806, “IEMA Identified”
- AR 01307329, “IEMA Identified Housekeeping Discrepancies”
- AR 01295867, “NRC Identified Fire Protection Equipment Staging Issue”
- AR 01309366, “ACE Needed For Activities in Containment & Drywell In Mode 3”
- AR 01304042, “NRC Identified Activities In Containment & Drywell During Mode 3”
- AR 01323352, “Air Void on Line 1RH117A”
- AR 01214706, “1DG01KC: Division 3 DG Emergency Stopped Due to Fuel Oil Leak”
- AR 01325494, “VC B Operability Run 9070.01 Flow Unsat”
- AR 01312043, “Air Was Observed During 9052.04 (RHR A to Fuel Pool)”
- AR 01272791, “Pressure Did Not Drop During Pass Panel Operability”
- AR 01237979, “NOS Identified Non-Safety Related Oil Installed in the HPCS Pump Motor”
- AR 01200237, “Division 2 SX Declare Operable Prior to EC 373594 Release for Operations”
- AR 01207130, “1MC078 Penetration Drainage Amount Inquiry”
- AR 01115664, “Evaluate GE Part 21 SLC Dilution Flow Applicability”
- AR 01179418, “Containment 825 Opening May Have Been Covered Prior to Mode 4”
- AR 01272803, “Pass Panel Operability Sample Cylinder Malfunction at 1PS02J”
- AR 01323827, “Potential Design Vulnerability: Single Open Phase”
- AR 01322414, “Fleet Review of Potential Design Vulnerability in Switchyard”

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

- CPS 2708.01, “Diesel Generator Voltage Regulator and Governor Test,” Revision1b
- CPS 2708.01D001, “Diesel Generator Voltage Regulator and Governor Test Data Sheet,” Revision 1
- CPS 3402.01P001, “Control Room HVAC (VC) Train Shifting,” Revision 3e
- CPS 3506.01P003, “Division 3 Diesel Generator Operations,” Revision 4c
- Prompt Investigation 1332224, “VC ‘A’ Chiller Failed to Load,” February 26, 2012
- WO 01347808-02, “OP-PMT 0SA01D Dryer,” March 12, 2012
- WO 01371198-03, “OP-PMT 0SA01D Dryer and Leak Check Rebuilt 0SA198B Valve,” March 12, 2012
- WO 01371199-03, “OP-PMT 0SA01D Dryer and Leak Check Rebuilt 0SA198A Valve,” March 12, 2012
- WO 01416877-02, “OP-PMT 0SA01D Dryer Perform Valve Functional Test,” March 12, 2012
- WO 01416878-02, “OP Return 0SA01D to Service and Perform Leak Check,” March 12, 2012
- WO 01517195-03, “0VC13CB Control Room HVAC Chiller B Troubleshooting”
- WO 01318718-01, “1PS038 Install Upgrade Valcor Valve (EC 379340)”
- WO 01337812-01, “Perform 9061.12R20 Primary Sampling Valve Position Verification Test”
- WO 01324596-01, "Replace / Bench Test / Adjust 1DO05A"
- WO 01278457-01, “Perform Thrust Verification and MOV Clean & Inspect 1SX012A”
- WO 01278458-01, “Perform Thrust Verification and MOV Clean & Inspect 1SX062A”
- E02-0VC99, “Control Room HVAC System (VC) Chiller Skid Control Panel 0VC13CB,” Sheet 55, Revision M
- E02-0VC99, “Control Room HVAC System (VC) Chiller Skid Control Panel 0VC13CB,” Sheet 57, Revision F

- AR 01335306, “NRC Questions on Evaluation of Risk & Use of Grace for LLRT”
- AR 01304015, “As-Left LLRT on 1PS038 Exceeds Administrative Leakage Limit”
- AR 01302723, “1PS038 and 1PS037 Will Not Stop Flow During Testing”
- AR 01326606, “1SX012A: No Inspection Ports on HBC”
- AR 01326617, “1SX012A: Beacon Grease on Geared Limit Switches”
- AR 01326929, “1SX062A: Beacon Grease in Geared Limit Switches”

1R20 Refueling and Other Outage Activities

- AR 01324867, “WHR: Request Apparent Cause Evaluation for 3 Individuals Violating WHR in C1R13”
- AR 01321716, “Instructor Assigned to 1CR13 LLRT Team WHR Issue”
- AR 01318109, “WHR Program Violation During C1R13”

1R22 Surveillance Testing

- MA-AA-723-301, “Periodic Inspection of Limitorque Model SMB/SB/SBD-000 Through 5 Motor Operated Valves,” Revision 7
- Clinton Nuclear Power Station Unit 1, “Inservice Testing Program Plan – Third Ten Year Interval,” Revision 0
- IST-CPS-BDOC-V-05, “Clinton Inservice Testing Program Bases Document – Diesel Fuel Oil,” Revision 4
- Issue Action Plan: IR 13087203/1308778, “Increase in Drywell Equipment Drain (RE) Inleakage,”
- IST Pump Evaluation Form, ER-AA-321 – Report No. 128, 1DO01PA, Revisions 0 and 1
- Operational and Technical Decision Making document 1049920-03, “Declining Performance Trend on Division 3 SX Pump (1SX01PC) Discharge Pressure,” June 3, 2010
- CPS 4001.01, “Reactor Coolant Leakage,” Revision 11
- CPS 3315.02, “Leak Detection (LD),” Revision 14c
- CPS 3402.01, “Control Room HVAC (VC),” revision 25c
- CPS 3402.01P001, “Control Room HVAC (VC) Train Shifting,” Revision 3e
- CPS 3506.01D003, “Diesel Generator 1C Operating Logs,” Revision 3a
- CPS 8451.04, “Limitorque Operator Removal/Installation,” Revision 14
- CPS 9053.04C001, “RHR Loop A Valve Operability,” Revision 2d
- CPS 9069.01, “Shutdown Service Water Operability Test,” Revision 47f
- CPS 9069.01D001, “Shutdown Service Water Operability Test Data Sheet,” Revision 45b
- CPS 9070.01, “Control Room HVAC Air Filter Package Operability Test Run,” Revision 27, Completion Date March 15, 2011
- CPS 9070.01D001, “Control Room HVAC Air Filter Package Operability Test Run Data Sheet,” Revision 25d, Completion Date March 15, 2011
- CPS 9080.14, “Diesel Generator 1C 24 Hour Run and Hot Restart – Operability,” Revision 37a
- CPS 9080.14D001, “Diesel Generator 1C 24 Hour Run and Hot Restart Data Sheet,” Revision 29c
- CPS 9080.12, “Diesel Generator Fuel Oil Transfer Pump Operability,” Revision 36a
- CPS 9170.01, “Control Room HVAC Chilled Water Pumps A, B Operability Test,” Revision 29b
- Control Room Logs for March 15, 2011
- WO 01297821, “IM Perform 1PISX018 Calibration (8801.01),” June 22, 2011
- WO 01366768, “Perform Thrust Verification and Clean/Inspect 1E12F028A,” March 1, 2012
- WO 01366768-03, “OPS PMT 1E12F028A – CPS 9053.04C001,” March 1, 2012
- WO 01455412-05, “OP 9069.01 SX Pump ‘C’ Operability Test,” October 11, 2011
- WO 01514502-01, “9070.01B21 Op Cntr RM M/U Air Filt FLW/HTR Oper-Trn B,”

- WO 01472818-01, “Obtain Oil Sample from Chiller for Analysis,”
- WO 01517342-01, “Section 8.9: Reactor Coolant System – Operational Leakage,”
- WO 01519173-01, “Section 8.9: Reactor Coolant System – Operational Leakage,”
- AR 01021241, “Late Scope Addition of 1B21F022C”
- AR 01264720, “1PISX018: Indicator Out Of Calibration At Top Of Range”
- AR 01275059, “VC ‘B’ HVAC SOW [System Outage Window] removed from WW [Work Week] 1142,”
- AR 01289621, “0VC13CB VC Chiller ‘B’ Head Pressure Controller Oscillating,”
- AR 01293245, “0VC13CB Control Room HVAC Chiller ‘B’ Condenser Pressure Cycling,”
- AR 01325494, “VC ‘B’ Operability Run 9070.01 Flow Unsat,”
- AR 01340319, “USAR Required Functional Test not Scheduled/Performed,”
- AR 01281802, “0VC13CB Freon: High Acid and Organic Matter Identified,”
- AR 01308778, “DW [Drywell] Equipment Drain Leakage Rising,”
- AR 01308203, “Increase in Drywell Equipment Drain (RE) Inleakage,”
- AR 01327424, “Inadequate MCR Crew Manning for Swings 2/15/2012”
- AR 01327496, “DG Logs and Room Tours Not Performed Hourly During 9080.14”
- AR 01328007, “EO Removed Wrong Tag During Temp Lift of a Clearance”
- AR 01328295, “NRC Questions Concerning DG DIV 3 Run Per 9080.14”

2RS1 Radiological Hazard Assessment and Exposure Controls

- RP-AA-301, “Radiological Air Sampling Program,” Revision 4
- RP-AA-800, “Control, Inventory, and Leak Testing of Radioactive Sources,” Revision 6
- RP-AA-460, “Controls for High and Locked High Radiation Areas,” Revision 21
- NOSA-CPS-11-06, “AR1120294; Radiation Protection Audit Report,” August 30, 2011
- RP-AA-700-1240, “Operation and Calibration of the Canberra ARGOS-5 Personnel Contamination Monitor,” Revision 0
- RP-AA-503, “Unconditional Release Survey Method,” Revision 5
- RP-AA-300, “Radiological Survey Program,” Revision 8
- RP-AA-203-1001, “Personnel Exposure Investigations,” Revision 6
- RP-AA-230-1001, Attachment 1, “Sample” Personnel Exposure Investigation,” Investigation 11-04-120
- RP-AA-230-1001, Attachment 1, “Sample” Personnel Exposure Investigation,” Investigation 11-04-121
- RP-AA-230-1001, Attachment 1, “Sample” Personnel Exposure Investigation,” Investigation 11-04-122
- RP-AA-210, “Dosimetry Issue, Usage, and Control,” Revision 22

2RS8 Radioactive Solid Waste Processing and Radioactive Material Handling, Storage, and Transportation

- Revision No. 25 of Certificate of Compliance No. 9228 for the Model No. 2000 Package, May 4, 2011
- MA-CL-716-104, “GE Model 2000 Transport Package Operations,” Revision 0
- MA-CL-716-104, “GE Model 2000 Transport Package Operations,” Revision 1
- RP-AA-600, “Radioactive/Waste Shipments,” Revision 12
- RP-AA-601, “Surveying Radioactive Material Shipments,” Revision 13
- RP-AA-602, “Packaging of Radioactive Material Shipments,” Revision 16
- Clinton Power Station, Unit No.1 – Issuance of Amendment RE: Request to Modify Facility Operating License in Support of the Use of Isotope Test Assemblies (TAC No. ME1643), June 15, 2010

4OA1 Performance Indicator Verification

- Nuclear Energy Institute 99-02, “Regulatory Assessment Performance Indicator Guideline,” Revision 6

4OA2 Identification and Resolution of Problems

- Event Notification (EN) #47489, “Actuation of the Reactor Protection System when the Reactor was Critical,” November 29, 2011
- EN #47587, “Invalid Actuation of General Containment Isolation Signals,” January 12, 2012
- LER 2011-004-00, “Automatic Reactor Scram During Removal of Main Generator”
- ER-AA-2003, “System Performance Monitoring and Analysis,” Revision 9
- ER-AA-2030, “Conduct of Plant Engineering,” Revision 12
- ER-AA-310, “Implementation of the Maintenance Rule,” Revision 8
- ER-AA-310-1004, “Maintenance Rule – Performance Monitoring,” Revision 8
- ER-AA-310-1007, “Maintenance Rule – Periodic (A)(3) Assessment,” Revision 4
- ER-AA-310-1008, “Exelon Maintenance Rule Process Map,” Revision 0
- LS-AA-125, “Corrective Action Program (CAP) Procedure,” Revision 15
- MA-AA-716-210, “Performance Centered Maintenance (PCM) Process,” Revision 13
- MA-AA-716-210-1001, “Performance Centered Maintenance (PCM) Templates,” Revision 9
- NO-AA-10, “Quality Assurance Topical Report,” Revision 86
- RM-AA-101-1008, “Processing and Storage of Records,” Revision 5.
- WC-AA-106, “Work Screening and Processing,” Revision 12
- 10 CFR 50.65(a)(3) Assessment of Maintenance Effectiveness, “Assessment Period: 3/1/2008 to 3/1/2010,” May 27, 2010
- Root Cause Report #1295617-05, “Automatic Scram on High Pressure During Approach to Unit Shutdown”
- Common Cause Evaluation AR 01309522, “Reactivity Management Performance Indicator in Variance”
- Common Cause Evaluation AR 01306419, “Work Delays Due to Operations Not Identifying Conflicts”
- Common Cause Evaluation AR 01308772, “Engineering Common Cause Analysis Performed on C1R13 Modification Revisions”
- Common Cause Evaluation AR 01260629, “Reactivity Management Performance”
- Common Cause Evaluation AR 01235674, “Perform a Common Cause Analysis on Operations Crew Performance and On Shift SOER 10-2 Observation Data Over the Past 17 Months”
- Common Cause Evaluation AR 01223868, “Common Cause Analysis on HU [Human Performance] Coded Issue Reports Generated On-Line”
- Common Cause Evaluation AR 01215260, “Common Cause Analysis of Diesel Generator System Issues”
- Common Cause Evaluation AR 01231102, “Perform Common Cause Evaluation on Work Hour Rule Program”
- Common Cause Evaluation AR 01240187, “Foreign Material Exclusion Program Trending Analysis Common Cause Evaluation”
- Common Cause Evaluation AR 01275199, “Common Cause Analysis on Operability, Reportability, or Functionality Issues by Operations Licensed Individuals”
- Apparent Cause Evaluation AR 1297701-06, “Inadvertent Actuation of Level 1 & 2 Logic”
- Apparent Cause Evaluation AR 01309488, “1TGCV1: ECR for TCV#1 [Turbine Control Valve #1] Oscillations and Impact to EH [Electro-Hydraulic] Pressure”
- 4th Quarter 2011 Coding & Analysis Report (AR 01283540),” February 24, 2012

- Clinton Power Station Updated Final Safety Analysis Report, Chapter 17, “Quality Assurance,” Revision 14
- ASME NQA-1, “Quality Assurance Program Requirements for Nuclear Facilities,” 1994, Supplement 17S-1, “Supplementary Requirements for Quality Assurance Records”
- Operational Decision Making (ODM) document 1122094, “Should a Scram from Low Power or Soft Shutdown be used to Start C1R13,” January 26, 2011
- Prompt Investigation (PINV) 1295617, “Reactor Scram During Turbine Trip,” November 30, 2011
- PINV 1297701, “Inadvertent Actuation of Level 1 & 2 Logic,” December 6, 2011
- NNOE 1295617-20-01, “Reactor SCRAM During Planned Plant Shutdown”
- ECR 0000403117, “1TGCV1 Valve Curve Out of Tolerance,” Revision 0
- EC 3872650, “Evaluate the TCV#1 Valve Curve Results,” Revision 0
- CPS 4001.02, “Automatic Isolation,” Revision 17a
- CPS 4001.02C001, “Automatic Isolation Checklist,” Revision 15d
- WO 01364404-30, “IM IAW EC383915 Remove TCCP EC 381213 – Division 1, Install EC 383951,” December 28, 2011
- AR 01112456, “1C85F0022: Deferral of PM for Steam Bypass Valve #1”
- AR 01117617, “Lack of PMs For Steam Bypass System Circuit Cards”
- AR 01182185, “Temperature and Humidity Levels in Vault Below Requirements”
- AR 01224290, “Documentation and Evaluation of Engineering Low Level HU Events”
- AR 01295617, “Reactor Scram During Turbine Trip”
- AR 01295710, “Turbine Bypass Valves Do Not Control Pressure”
- AR 01295793, “Feedwater Level Control System Response to Scram on 11/29/11”
- AR 01295807, “1PL89JA-32TG1A: Main Generator Did Not Trip On Reverse Power”
- AR 01296813, “4.0 Critique Following Reactor Scram for C1R13”
- AR 01297648, “Division 1 NSPS Power Monitor Card Clamshells Cracked”
- AR 01297701, “Inadvertent Actuation of Level 1 & 2 Logic”
- AR 01297798, “Main Steam Line Plugs Air Pressure”
- AR 01319008, “C1R13 Lessons Learned Outage Lesson Learned Plant Engineering Turbine Reset”
- AR 01333977, “CAPR 1295617-33 Improperly Closed to Service Request”
- AR 01275199, “Trend Identified in Operability and Functionality Reviews”
- AR 01316710, “Temp-Humidity Levels in Records Management Vault Outside of Requirements”
- AR 01325373, “Temp-Humidity Levels in Records Management Vault Outside of Requirements”
- AR 01311426, “Temp-Humidity Levels in Records Management Vault Outside of Requirements”
- AR 01301139, “Humidity Level in Records Management Vault Exceeds Procedural Requirement”
- AR 01249320, “Temp/Humidity in Primary Vault Out of Spec”
- AR 01241784, “Temp/Humidity Level in Vault Out of Spec”
- AR 01236670, “Humidity Level in Vault Out of Spec”
- AR 01328272, “Humidity Levels in Records Management Vault Outside of Requirements”
- AR 01328289, “0VS28M: Red Light Is On Humidifier Fault Lamp (EIN 0ZLVS028)”
- AR 01265354, “Humidity in Primary Vault Out of Spec”
- AR 01309488, “1TGCV1: ECR for TCV#1 Oscillations and Impact to EH Pressure”
- AR 01306503, “EHC/CV Oscillations at ~ 80 Power”
- AR 01304146, “Circulating Water Level Floats Partially Failed 3813.01 Section 8.1.3”
- AR 01324420, “Gaps in Engineering Technical Evaluations”

- AR 01316386, “1H13P653: Received MCR [Main Control Room] Annunciator 5006-3G RC&IS [Rod Control and Information System] Inoperable”
- AR 01304798, “Received MCR Annunciator 5006-3G RC&IS Inoperable”
- AR 01319000, “1H13P653: Received MCR Annunciator 5006-3G RC&IS Inoperable”
- AR 01319035, “Degrading Trend – RC&IS Lockups”
- AR 01321963, “Benchmarking for Operability and Functionality Reviews”
- AR 00974191, “Perform Common Cause Evaluation of Operability Evaluation Related IRs”
- AR 01010995, “Apparent Cause Evaluation to Investigate Organizational Weaknesses”
- AR 01275199, “Trend Identified in Operability and Functionality Reviews”

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

- LER 05000461/2011-005-00, “Missed Surveillance Due To Preconditioning Valve Prior To Leak Rate Test,” January 24, 2012
- LER 05000461/2011-006-00, "Condition Prohibited By Technical Specifications Due To Failed Missed Surveillance"
- EN #47544, “H2 Leak from Main Generator and NOUE Declared,” December 21, 2011
- Apparent Cause Evaluation 1305735-25, “Significant H2 Leak From Main Generator”
- Common Cause Analysis (CCA) #1228140, “Common Cause Analysis of Hydrogen Leak Issues”
- ODM document 1308464, “Main Generator T-4 Neutral Bushing has a 14 drop per day Viscasil Leak Following Repairs to the Bushing in C1R13”
- NNOE 1305735-22-01, “Main Generator Hydrogen Loss”
- EC 387315, “Evaluate Past Operability of 1E51-F040 LLRT Failure,” Revision 0
- EC 387261, “Evaluate Past Operability of 1E12-F042C LLRT Failure,” Revision 0
- CPS 9861.05D010, “RCIC Turbine Exhaust Water Leak Rate Test Data Sheet (S-MC039k12),” Revision 24
- AR 01228140, “Trend in Hydrogen Leaks”
- AR 01304348, “Generator Manway Bolts Leaking During Leak Snoop”
- AR 01305735, “Significant H2 Leak From Main Generator”
- AR 01305937, “Leak Identified on HY Cooler 1TG02AD During Air Test”
- AR 01307258, “Timeliness of NRC ENS Notification During NOUE”
- AR 01308930, “1MP01K: Formal Investigation Needed For Main Generator Leaks”
- AR 01317663, “1MP01K: Main Generator Daily Hydrogen Use Increase”
- AR 01300655, “NRC Identified Question of LLRT of 1E51-F040”
- AR 01310612, “Unacceptable Preconditioning Identified For 1E51F040”
- AR 01305725, “NRC ID Assess 1E12-F042C LLRT Failure for Operability/Reportability”
- AR 01299275, “1E12-F042C Failed Leak Rate Testing”
- AR 01300566, “1E12-F042C Inspection Results”
- AR 01202456, “NRC Question on RCS PIV Surveillance Testing”

4OA5 Other Activities

- CPS-11-049, “ASME Requirements, Reactor Overfill Analysis, Spurious HPCS Operation,” November 10, 2011
- EC 386645, “Reactor Overfill Impact on Main Steam Piping and SRVs Due to Fire Induced Spurious HPCS Operation,” Revision 0
- IP-M-0792, “MSRV Discharge Transient Force Analysis for Reactor Overfill,” Revision 0
- IP-M-0793, “Piping Stress Analysis for SRV Discharge Transient Forces,” Revision 0

LIST OF ACRONYMS USED

ADAMS	Agencywide Documents and Management System
ASME	American Society of Mechanical Engineers
AR	Action Request
AVR	Automatic Voltage Regulator
BPV	Bypass Valve
BVD	Bypass Valve Demand
BI	Barrier Integrity
BVD	Bypass Valve Demand
C1R12	Unit 1 Refueling Cycle 12
C1R13	Unit 1 Refueling Cycle 13
CAP	Corrective Action Program
CFR	Code of Federal Regulations
CPS	Clinton Power Station
CST	Central Standard Time
CW	Circulating Water
DC	Direct Current
DG	Diesel Generator
EC	Engineering Change
ECR	Engineering Change Request
ED	Electronic Dosimeter
EH	Electro-hydraulic
EHC	Electro-hydraulic Control
FIN	Finding
gpm	Gallons-per-minute
H2	Hydrogen
HPCS	High Pressure Core Spray
HVAC	Heating, Ventilation, and Air Conditioning
IE	Initiating Events
IMC	Inspection Manual Chapter
INPO	Institute of Nuclear Power Operations
IP	Inspection Procedure
ISI	Inservice Inspection
IST	Inservice Testing
LER	Licensee Event Report
LOCA	Loss-of-Coolant Accident
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
MOV	Motor Operated Valve
MS	Mitigating Systems
MSRV	Main Steam Relief Valve
NCV	Non-Cited Violation
NOUE	Notice of Unusual Event
NRC	U.S. Nuclear Regulatory Commission
OM	Operations and Maintenance
PARS	Publicly Available Records System
PINV	Prompt Investigation
PIV	Pressure Isolation Valve
PM	Preventive Maintenance
PMCEI	Plant Material Condition Excellence Initiative

psig	Pounds-per-square-inch-gauge
RCS	Reactor Coolant System
RC&IS	Rod Control & Information System
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RWP	Radiation Work Permits
SBPC	Steam Bypass and Pressure Control
SDP	Significance Determination Process
SRV	Safety Relief Valve
SSC	Structure, System, and Component
SX	Shutdown Service Water
TCV	Turbine Control Valve
TS	Technical Specification
TSSR	Technical Specification Surveillance Requirement
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
WANO	World Association of Nuclear Operators
WO	Work Order

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Sincerely,

/RA/

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

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