April 19, 2007

Mr. Christopher M. Crane
President and Chief Nuclear Officer
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: CLINTON POWER STATION NRC INTEGRATED INSPECTION REPORT 05000461/2007002

Dear Mr. Crane:

On March 31, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed an integrated inspection at your Clinton Power Station. The enclosed report documents the inspection results, which were discussed on April 5, 2007, with Mr. Bryan Hanson and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and to 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, two self-revealing findings of very low safety significance (Green), one of which was determined to involve a violation of NRC requirements, were identified. However, because of the very low safety significance and because it is entered into your corrective action program, the NRC is treating this finding as a non-cited violation (NCV) consistent with Section VI.A of the NRC Enforcement Policy.

If you contest any NCV in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the US Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, US Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at Clinton Power Station facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure, and your response (if any) will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), 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, Chief
Branch 1
Division of Reactor Projects

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

Enclosure: Inspection Report No. 05000461/2007002
w/Attachment: Supplemental Information

cc w/encl: Site Vice President - Clinton Power Station
Plant Manager - Clinton Power Station
Regulatory Assurance Manager - Clinton Power Station
Chief Operating Officer
Senior Vice President - Nuclear Services
Vice President - Operations Support
Vice President - Licensing and Regulatory Affairs
Manager Licensing - Clinton Power Station
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
Chairman, Illinois Commerce Commission

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Vice President - Licensing and Regulatory Affairs
Manager Licensing - Clinton Power Station
Senior Counsel, Nuclear, Mid-West Regional Operating Group
Document Control Desk - Licensing
Assistant Attorney General
Illinois Emergency Management Agency
State Liaison Officer, State of Illinois
Chairman, Illinois Commerce Commission

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-461
License No:	NPF-62
Report No:	05000461/2007002
Licensee:	AmerGen Energy Company, LLC
Facility:	Clinton Power Station
Location:	Clinton, IL 61727
Dates:	January 1 through March 31, 2007
Inspectors:	B. C. Dickson, Senior Resident Inspector D. Tharp, Resident Inspector R. Winter, Reactor Engineer
Approved by:	Mark Ring, Chief Branch 1 Division of Reactor Projects

Enclosure

SUMMARY OF FINDINGS

IR 05000461/2007002; AmerGen Energy Company LLC; 01/01/52007 - 03/31/2007; Clinton Power Station; Event Followup.

This report covers a 3-month period of baseline resident inspection and announced baseline inspections on radiation protection and security. The inspection was conducted by Region III inspectors and the resident inspectors. Two Green findings 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 sate operation of commercial nuclear power reactors is described in NUREG-1649, “Reactor Oversight Process,” Revision 3, dated July 2000.

A. Inspector-Identified and Self Revealing Findings

Cornerstone: Mitigating Systems

Green: A finding of very low safety significance (Green) involving a Non-Cited Violation of 10 CFR 50, Appendix B, Criterion V, was self revealed when a low main condenser vacuum alarm was received in the main control room. The alarm was caused by the failure of an electronic circuit card. This circuit card failure also resulted in the main turbine bypass valves being interlocked closed (loss of safety function). The inspectors determined that the cause of this issue was inadequate instructions contained in the licensee’s Performance Centered Maintenance (PCM) process.

The finding was greater than minor because failure to have adequate instructions to implement an effective preventive maintenance program could be reasonably viewed as a precursor to a more significant event. Additionally, this finding could affect the mitigating systems cornerstone in that it is associated with a degraded condition that could concurrently influence mitigation equipment and the operators’ response to an initiating event. This finding was of very low safety significance because the exposure time was of short duration, less than 3 days. (Section 4OA3.1)

Green: A finding of very low safety significance (Green) was self-revealed following the loss of the division 3 shutdown service water (SX) system on August 17, 2006. The loss of division 3 of SX occurred when a security guard bumped an SX circuit breaker hand switch for the cross tie valve, 1SX014C, with a piece of protective equipment. This finding resulted from the licensee’s failure to do an adequate inadvertent contact configuration control risk assessment during the implementation of a 2005 requirement for security personnel to carry new equipment on their person.

The finding was more than minor because it impacted the mitigating systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. With the circuit breaker in the OFF position, 1SX014C would remain open during a loss of offsite power event. In this configuration, the

SX system could not perform its safety function of supplying cooling water to both the division 3 diesel generator and the high pressure core spray pump room cooling system. This finding was of very low safety significance due to the short duration exposure time, less than three days, and credit for operator actions to restore the system back to service. This finding affected the work practices component of the cross-cutting area of human performance. Licensee management failed to ensure the proper management and oversight of security personnel rounds activities. (Section 4OA3.2)

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

No findings of significance were identified.

REPORT DETAILS

Summary of Plant Status

The plant was operated at approximately 96 to 97 percent rated thermal power (maintaining 100 percent electrical output) throughout the inspection period with one exception. On March 2, 2007, operators lowered reactor power to approximately 50 percent to make repairs to an extraction steam line leak. Operators returned reactor power to 96 percent following the completion of repairs on March 4, 2007, and remained there through the close of the inspection period.

1. REACTOR SAFETY

Cornerstone: Initiating Events, Mitigating Systems, and Barrier Integrity

1R04 Equipment Alignments (71111.04Q)

a. Inspection Scope

The inspectors performed partial walkdowns of accessible portions of divisions of risk-significant mitigating systems equipment during times when the divisions were of increased importance due to the redundant divisions or other related equipment being unavailable. The inspectors utilized the valve and electric breaker checklists listed at the end of this report to verify that the components were properly positioned and that support systems were lined up as needed. The inspectors also examined the material condition of the components and observed operating parameters of equipment to verify that there were no obvious deficiencies. The inspectors reviewed outstanding work orders and issue reports (IRs) associated with the divisions to verify that those documents did not reveal issues that could affect division function. The inspectors used the information in the appropriate sections of the Updated Safety Analysis Report (USAR) to determine the functional requirements of the systems. The documents listed at the end of this report were also used by the inspectors to evaluate this area.

The inspectors performed four samples by verifying the alignment of the following divisions:

- Low pressure core spray system;
- High pressure core spray system during reactor core isolation cooling system testing;
- Residual heat removal A during the system outage window for residual heat removal B train and
- Reactor core isolation cooling during the high pressure core spray system outage window.

b. Findings

No findings of significance were identified.

1R05 Fire Protection (71111.05Q)

a. Inspection Scope

The inspectors conducted fire protection walkdowns which were focused on availability, accessibility, and the condition of fire fighting equipment, the control of transient combustibles and ignition sources, and on the condition and operating status of installed fire barriers. The inspectors selected fire areas for inspection based on their overall contribution to internal fire risk, as documented in the individual plant examination of external events with later additional insights, and their potential to impact equipment which could cause a plant transient, to verify that fire hoses and extinguishers were in their designated locations and available for immediate use, that fire detectors and sprinklers were not obstructed, that transient material loading was within the analyzed limits, and that fire doors, dampers, and penetration seals appeared to be in satisfactory condition. The inspectors verified that minor issues identified during the inspection were entered into the licensee's corrective action program.

The inspectors reviewed portions of the licensee's fire protection evaluation report and the USAR to verify consistency in the documented analysis with installed fire protection equipment at the station.

The inspectors completed nine samples by inspection of the following areas:

- Fire Zones CB-2 Division 2 and CB-4 Division 1 cable spreading room;
- Fire Zone T-1m, Turbine deck;
- Fire Area C-2, 737'-0" Containment;
- Fire Zones A-2a, A-1a, and A-1b Reactor core isolation cooling pump room and auxiliary building general access areas;
- Fire Zone R-1c, RW 702 General access area;
- Fire Zones A-1a and A-1b Auxiliary building general access area;
- Fire Zone D-5a Division 1 diesel generator room;
- Fire Zone T-1a, Turbine building 712' general access area and
- Fire Area C-2, 803'-3," 789'-1," 778', and 755' Containment.

b. Findings

No findings of significance were identified.

1R11 Licensed Operator Requalification (71111.11)

.1 Quarterly Resident Inspector Review

a. Inspection Scope

The inspectors reviewed licensed-operator requalification training to evaluate operator performance in mitigating the consequences of a simulated event, particularly in the areas of human performance. The inspectors evaluated operator performance attributes which included communication clarity and formality, timely performance of appropriate operator actions, appropriate alarm response, proper procedure use and adherence, and senior reactor operator oversight and command and control.

Crew performance in these areas was compared to licensee management expectations and guidelines as presented in the following documents:

- SE-LOR-31, "Loss of coolant accident, blowdown, and reflood," Revision 1;
- OP-AA-101-111, "Roles and responsibilities of on-shift personnel," Revision 1;
- OP-AA-103-102, "Watchstanding practices," Revision 6;
- OP-AA-104-101, "Communications," Revision 1, and
- OP-AA-106-101, "Significant event reporting," Revision 7.

The inspectors also assessed the performance of the training staff evaluations involved in the requalification process. For any weaknesses identified, the inspectors observed that the licensee evaluators also noted the issues and discussed them in the critique at the end of the session. The inspectors verified all issues were captured in the training program and licensee corrective action process.

These activities completed one inspection sample.

b. Findings

No findings of significance were identified.

1R12 Maintenance Effectiveness (71111.12)

.1 Routine Evaluation

a. Inspection Scope

The inspectors reviewed the effectiveness of the licensee's maintenance efforts in implementing 10 CFR 50.65 (the maintenance rule (MR)) requirements, including a review of scoping, goal-setting, performance monitoring, short and long-term corrective actions, and current equipment performance problems. These systems were selected based on their designation as risk-significant under the maintenance rule, or being in the increased monitoring (MR category (a)(1)) group. In addition, the inspectors interviewed the system engineers and maintenance rule coordinator. The inspectors also reviewed IRs and associated documents for appropriate identification of problems, entry into the corrective action system, and appropriateness of planned or completed actions. The documents reviewed are listed at the end of the report. The inspectors completed one sample by reviewing the following:

- Condensate booster pump and associated high risk air operated valves.

b. Findings

No findings of significance were identified.

.2 Periodic Evaluation

a. Inspection Scope

The inspectors examined the maintenance rule periodic evaluation report completed for the period of March 1, 2004, through March 1, 2006. The inspectors reviewed a sample of (a)(1) Action Plans, Performance Criteria, Functional Failures, and Condition Reports to evaluate the effectiveness of (a)(1) and (a)(2) activities. These same documents were reviewed to verify that the threshold for identification of problems was at an appropriate level and the associated corrective actions were appropriate. Also, the inspectors reviewed the maintenance rule procedures and processes. The inspectors focused the inspection on the following systems:

- Control Room Ventilation (VC);
- Diesel Generator (DG);
- Reactor Core Isolation Cooling (RI);
- Neutron Monitoring (NR); and
- Leak Detection (LD).

The inspectors verified that the periodic evaluation was completed within the time restraints defined in 10 CFR 50.65 (once per refueling cycle, not to exceed 24 months). The inspectors also ensured that the licensee reviewed its goals, monitored Structures, Systems, and Components (SSCs) performance, reviewed industry operating experience, and made appropriate adjustments to the maintenance rule program as a result of the above activities.

The inspectors verified that:

- the licensee balanced reliability and unavailability during the previous cycle, including a review of high safety significant SSCs;
- (a)(1) goals were met, that corrective action was appropriate to correct the defective condition, including the use of industry operating experience, and that (a)(1) activities and related goals were adjusted as needed; and
- the licensee has established (a)(2) performance criteria, examined any SSCs that failed to meet their performance criteria, and reviewed any SSCs that have suffered repeated maintenance preventable functional failures including a verification that failed SSCs were considered for (a)(1).

In addition, the inspectors reviewed maintenance rule self-assessments and audit reports that addressed the maintenance rule program implementation.

This review represented five triennial inspection samples.

b. Findings

No findings of significance were identified.

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

a. Inspection Scope

The inspectors observed the licensee's risk assessment processes and considerations used to plan and schedule maintenance activities on safety-related structures, systems, and components particularly to ensure that maintenance risk and emergent work contingencies had been identified and resolved. The inspectors completed seven samples by assessing the effectiveness of risk management activities for the following work activities or work weeks:

- High pressure core spray pump maintenance and surveillance activities;
- Low pressure core spray pump and water leg pump maintenance and operability surveillance activities;
- Risk assessment of for both division 2 main control room ventilation and standby gas treatment system planned maintenance activities;
- Risk assessments for anticipated transient without scram (Reactor pressure), and main steam line tunnel channel calibrations;
- Risk assessment and planned work activities for division 1 main control room ventilation and standby gas treatment system maintenance windows;
- Plans and risk assessment for CPS 9071.09, CO2 Puff test, following 0CO602 maintenance, and
- Risk assessment and planned work activities for residual heat removal "B" and "C" work windows.

b. Findings

No findings of significance were identified.

1R15 Operability Evaluations (71111.15)

a. Inspection Scope

The inspectors reviewed the following operability determinations and evaluations affecting mitigating systems to determine whether operability was properly justified and the component or system remained available such that no unrecognized risk increase had occurred. The inspectors completed two samples of operability determinations and evaluations by reviewing the following:

- Operability basis for IR 582162 and IR 582173, division 2 and 3 diesel generator air intake filter housing and
- Operability basis for IR 589461 and EC 334968, reactor core isolation cooling oil level low.

b. Findings

No findings of significance were identified.

1R19 Post Maintenance Testing (71111.19)

a. Inspection Scope

The inspectors reviewed the post maintenance testing activities associated with maintenance or modification of important mitigating, barrier integrity, and support systems that were identified as risk significant in the licensee's risk analysis. The inspectors reviewed these activities to verify that the post maintenance testing was performed adequately, demonstrated that the maintenance was successful, and that operability was restored. During this inspection activity, the inspectors interviewed maintenance and engineering department personnel and reviewed the completed post maintenance testing documentation. The inspectors used the appropriate sections of the Technical Specifications (TS) and USAR, as well as the documents listed at the end of this report, to evaluate this area.

Testing subsequent to the following activities was observed and evaluated to complete four inspection samples:

- Work orders 995047 and 995048, addressing loose bolts on division 3 diesel generator;
- Division 1 main control room ventilation addressing hydramotor 0FZC024 and calibration of control room chiller inlet temperature;
- Standby gas treatment monthly operability surveillance to assess operability following planned maintenance activities, and
- Work activities related to the electrochemical probe in the residual heat removal system heat exchanger and subsequent residual heat removal "B" operability surveillance.

b. Findings

No findings of significance were identified.

1R22 Surveillance Testing (71111.22)

a. Inspection Scope

The inspectors witnessed selected surveillance testing and/or reviewed test data to verify that the equipment tested using the surveillance procedures met the TS, the Technical Requirements Manual (TRM), the USAR, and licensee procedural requirements, and demonstrated that the equipment was capable of performing its intended safety functions. The activities were selected based on their importance in verifying mitigating systems capability and barrier integrity. The inspectors used the documents listed at the end of this report to verify that the testing met the frequency requirements; that the tests were conducted in accordance with the procedures, including establishing the proper plant conditions and prerequisites; that the test acceptance criteria were met; and that the results of the tests were properly reviewed and recorded. In addition, the inspectors interviewed operations, maintenance and engineering department personnel regarding the tests and test results.

The inspectors evaluated the following surveillance tests to complete seven inspection samples:

- High pressure core spray quarterly operability surveillance test;
- Low pressure core spray pump and water leg pump operability surveillance;
- Division 2 diesel generator surveillance test;
- Division 3 diesel generator monthly surveillance test;
- Anticipated transient without scram, reactor level channel check, calibration and functional test;
- Residual heat removal "B" operability surveillance, and
- Division 1 diesel generator operability surveillance test.

These inspections represented the completion of six in-service tests and one routine test.

b. Findings

No findings of significance were identified.

1R23 Temporary Plant Modifications (71111.23)

a. Inspection Scope

The inspectors reviewed and evaluated the following temporary plant modification on risk significant equipment to verify that the instructions were consistent with applicable design modification documents and that the modifications did not adversely impact system operability or availability. The inspectors interviewed operations, engineering and maintenance personnel as appropriate and reviewed the design modification documents and the 10 CFR 50.59 evaluations against the applicable portions of the USAR. The documents listed at the end of this report were also used by the inspectors to evaluate this area. The inspectors reviewed the issues that the licensee entered into its corrective action program to verify that identified temporary modification problems were being entered into the program with the appropriate characterization and significance. The inspectors also reviewed the licensee's corrective actions for temporary modification related issues documented in selected condition reports. The condition reports are specified in the List of Documents Reviewed.

The inspectors completed one inspection sample by reviewing the following temporary modification:

- EC 365082, Rev 0, Disconnect leads for circuit breaker 252-AT1AA1 relay 74-AT1AA1 to bypass alarm to window 1D on main control room panel 1H13-P877-14A.

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES (OA)

4OA1 Performance Indicator Verification (71151)

Cornerstones: Initiating Events

a. Inspection Scope

The inspectors sampled the licensee's submittals for performance indicators for the period of first quarter 2005 through fourth quarter 2006. The inspectors used performance indicator definitions and guidance contained in Revision 4 of Nuclear Energy Institute (NEI) document 99-02, "Regulatory Assessment Performance Indicator Guideline" to verify the accuracy of the performance indicator data. The inspectors performed three samples by reviewing the following:

- Scrams with Loss of Normal Heat Removal;
- Unplanned Scrams per 7,000 Critical Hours; and
- Unplanned Power Changes per 7,000 Critical Hours.

b. Issues and Findings

No findings of significance were identified.

4OA2 Identification and Resolution of Problems (71152)

.1 Routine Review and Identification 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 system at an appropriate threshold, that adequate attention was being given to timely corrective actions, and that adverse trends were identified and addressed. Minor issues entered into the licensee's corrective action system as a result of inspectors' observations are generally denoted in the report.

b. Findings

No findings of significance were identified.

.2 Reactor water clean-up system leakage (Annual sample)

a. Inspection Scope

The inspectors reviewed licensee actions associated with recurring alarms for reactor water clean up 'A' heat exchanger room delta temperature and an increasing reactor water clean up differential flow, to determine if corrective actions were timely, accurate and complete. The inspectors interviewed licensee personnel from operations and work management and reviewed issue reports and work documents to complete one inspection sample. A list of documents reviewed is included in the attachment to this report.

b. Observations

On January 23, 2007, main control room operators responded to annunciators 5004-3F, SPDS CSF Alarm due to secondary containment delta temperature and 5000-3F, Reactor water cleanup equipment room differential temperature high. Operators also noticed an increase in reactor water cleanup differential flow associated with these alarms, and entered the reactor coolant leakage off-normal procedure, CPS 4001.01. Plant personnel then made an entry into the reactor water cleanup A heat exchanger room, but observed no indications of system breach or gross leakage. An issue report was generated and operators continued to monitor the conditions of the reactor water cleanup A heat exchanger room. Between January 23 and February 5, 2007, operators responded to these same alarms several times, and conducted a historical search on this issue. Between September 24, 2004, and December 7, 2006, nine issue reports had been generated to address this issue. On February 2, 2007, the maintenance organization identified high tailpipe temperatures downstream of reactor water cleanup regenerative heat exchanger AA shell side drain valves 1G33-F018A and 1G33-F019A. In addition, using thermography, maintenance technicians confirmed high tailpipe temperatures and ongoing seat leakage past regenerative heat exchanger AA tube side drain valves 1G33-F014A and 1G33-F015A, which had been identified in 2004. On February 5, 2007, operations swapped reactor water cleanup heat exchanger trains to place the B train in service. All of the issue reports related to this drain valve leak by and alarm responses were closed to work order 992459. The inspectors looked at work order 992459 to ensure all of the issues were captured for repair, but the work order was still in the “plan” stage, and no job steps had been written. The inspectors questioned the corrective action program manager and a work week manager about the process of converting corrective action program documents into work documents. These individuals informed the inspectors that in the process, all of the items closed to the work order were linked through references in Passport, and the maintenance planner had access to all of the issue reports. It was the maintenance planner’s responsibility to ensure the work order included job steps to address all of the deficiencies identified through the corrective action process.

c. Conclusions

The inspectors did not identify any performance deficiencies in the licensee’s response to the reactor water cleanup heat exchanger valve leakage and resultant alarms. The licensee did identify the source of leakage and cause for the alarms and addressed each of these appropriately in its corrective action program. The inspectors concluded that the licensee’s process for converting corrective action program items to work documents had some vulnerability however. Once issue reports are closed to a work order, a single maintenance planner has responsibility to correctly identify, classify, and develop actions to correct all of the conditions addressed in multiple issue reports. The feedback process incorporated in the corrective action program notifies the originator of the issue that "work" will be done to address the issue. The inspectors were concerned that some issues, addressed through the licensee's corrective action program and closed to the work process, could get missed or dropped, and the organization would not be aware that the anticipated actions were not taken. The inspectors discussed their concern with the licensee.

.3 Adverse effects of scaffolding on safety related equipment (Annual sample)

a. Inspection Scope

The inspectors reviewed licensee corrective action program documents and work documents related to scaffolding between March 1, 2005, and February 28, 2007, to identify the possible effects of scaffolding on the availability or operability of safety related systems. A list of documents reviewed is included in the attachment to this report. This activity represented one inspection sample.

b. Observations

A word search of the licensee's corrective action program for "scaffolding" between March 2005 and February 2007 resulted in a thirty eight page list of documents. Many of these 'hits' were related to work requiring scaffolding or requesting scaffolding be built or wrecked. Only seven of these issues were related to improperly built or evaluated scaffolding. The inspectors reviewed these issues and had no concerns that scaffolding could have caused a safety function or safety related system to be inoperable or unavailable as described in the industry operating experience reviewed.

4OA3 Event Followup (71153)

.1 (Closed) LER 05000461/2006002-00 and 01. Main Turbine Bypass System Safety Function Lost Due to Circuit Card Failure

Introduction: A finding of very low safety significance (Green) involving a non-cited violation of 10 CFR 50, Appendix B, Criterion V was self-revealed when a low main condenser vacuum alarm was received in the main control room. The alarm was caused by the failure of an electronic circuit card. This card failure also resulted in the main turbine bypass valves being interlocked closed (loss of safety function). The inspectors determined that the cause of this issue was inadequate instructions contained in the licensee's Performance Centered Maintenance (PCM) process.

Description: On August 17, 2006, operators in the control room received a main condenser low vacuum alarm. The plant was at full power operations. After verifying all pertinent indications were normal, the licensee identified that with this alarm in place, the main turbine bypass valves were interlocked closed. Clinton Improved TS require the main turbine bypass valves to be operable when the reactor thermal power is greater than or equal to 21.6 percent. The operators correctly entered TS 3.7.6, "Main Turbine Bypass System," which has a limiting condition for operation (LCO) 2-hour action statement.

Shortly after entry into this LCO action statement, a troubleshooting team determined that the failure of a single main condenser pressure trip unit was the cause of the event. The alarm/trip units (electronic circuit cards) are arranged in a 1 out of 2 logic to inhibit the main turbine bypass valve opening function when main condenser vacuum is too low for steam to be admitted. The licensee performed a system modification that involved lifting the leads to the electronic circuit card to remove the low vacuum inhibit logic input. This action restored the main turbine bypass valve safety function. The low vacuum light extinguished as expected.

As part of the investigation into this event, the licensee completed an equipment apparent cause evaluation (EACE), to find the apparent cause of the low vacuum alarm and subsequent loss of safety function. A failure analysis completed by the vendor showed that the circuit card had degraded due a leaky electrolytic capacitor. The licensee's EACE identified that the electronic circuit card was approximately 27 years old and had approximately 22 years of actual service time.

The licensee's PCM process, as described in licensee procedure MA-AA-716-210, is a process for selecting effective preventive maintenance (PM) activities for components to minimize consequential failures. The PCM provides standardized generic PM tasks and frequencies based on the importance of the component. The results of the PCM process are PCM Templates which contain recommended PM frequencies for critical components. According to the licensee, Electric Power Research Institute's Plant Material Condition Excellence Initiative (PMCEI) was the basis document for the PCM process. A licensee review of the PCM template applicable to this electronic circuit card was performed. The PCM Template, "Circuit Cards - Moore Industries"' contains a specific section for the model direct current alarm (DCA) card that has a requirement to replace it every 30 years.

The licensee investigation concluded that the cause of this event was the failure of the electrolytic capacitor on the main condenser pressure trip unit due to the PCM template not specifying the appropriate replacement interval for Moore Industries model DCA cards. The circuit card replacement frequency in the PCM template for Moore model DCA cards is 30 years, and is in direct contrast to a PCM template created for power supplies, which addresses electrolytic capacitor aging, and has a replacement frequency of 7.5 years. The PMCEI basis document states "A lifetime for a particular card was calculated by identifying all the sub-components on that card and then determining the individual failure rates for the sub-components. The sub-component failure rates were taken from published databases. An overall circuit card failure rate was then determined based on the sub-component failure rates." Since the Moore Industries DCA cards had electrolytic capacitors as sub-components, the cards should have been replaced on a 7.5 year frequency, not the 30 year frequency in the PCM Template.

The inspectors reviewed procedures MA-AA-716-210, "Performance Centered Maintenance Process," and MA-AA-716-210-1001, "Performance Centered Maintenance Templates" to ensure that the guidance contained in the basis document was incorporated into the process procedures. Specifically, the inspectors reviewed step 4.13 of MA-AA-716-210 which describes the process of making changes to existing PCM templates or creating new PCM templates. During this review the inspectors concluded that the procedure lacks specific guidance that would ensure that individuals reviewing, changing, or creating a new PCM template would be aware of the PMCEI document statement referenced above. The inspectors concluded that this lack of guidance was the cause of the PCM template for the Moore model DCA card being incorrect.

Analysis: Failure of the electrical circuit card resulted in a loss of safety function for the main turbine bypass valves. The PCM template contained an incorrect PM frequency, which resulted in an unexpected failure of the electrical circuit card. This issue was caused by inadequate instructions in the licensee PCM program process procedure. This was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in accordance with Inspection Manual Chapter (IMC) 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on November 16, 2006. Failure to have instructions to implement an effective preventive maintenance program could be reasonably viewed as a precursor to a more significant event. Additionally, this finding could affect the mitigating systems cornerstone in that it is associated with a degraded condition that could concurrently influence mitigation equipment and the operators' response to an initiating event (Attribute: External Factor - Loss of Heat Sink).

The inspectors completed a phase 1 significance determination of this issue using IMC 0609, "Significance Determination Process," Appendix A, Attachment 1, dated March 23, 2007. The inspectors concluded that the finding affected the mitigating systems cornerstone. The inspectors answered "Yes" to question 2, which asked whether this issue resulted in a loss of system safety function. The inspectors completed a phase 2 analysis and determined that this issue was of very low safety significance (Green) because the exposure time was of short duration, less than three days.

Enforcement: 10 CFR 50 Appendix B, Criterion V, states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Contrary to the above, the licensee failed to develop an adequate PCM template because of lack of sufficient details in procedure MA-AA-716-210, "Performance Centered Maintenance Program." This failure resulted in loss of the safety function of the main turbine bypass valves on August 17, 2006. This was a violation. This issue was considered to be of very low safety significance and was entered into the licensee's corrective action program as IR 555579. The licensee performed an apparent cause evaluation that developed actions to correct this issue. Therefore, this issue is being treated as non-cited violation **(NCV 05000461/2007-002-01)**. Licensee Event Reports 05000461/2006002-00 and 01, "Main Turbine Bypass System Safety Function Lost Due to Circuit Card Failure" are closed.

.2 (Closed) LER 05000461/2006004-00: Inadequate Configuration Control Risk Assessment Causes Loss of Safety Function

Introduction: A finding of very low safety significance (Green) was self-revealed following the loss of the division 3 shutdown service water (SX) on August 17, 2006. This finding resulted from the licensee's failure to do an adequate inadvertent contact configuration control risk assessment during the implementation of a 2005 requirement for security personnel to carry new equipment on their person.

Description: During the performance of shiftly rounds, a security officer bumped a circuit breaker hand switch for the SX to normal service water crosstie isolation valve (1SX014C) to the OFF position. After the breaker hand switch was bumped, a high pressure core spray system alarm was received in the main control room coincident with the loss of position indication for 1SX014C. With the breaker in the OFF position, the division 3 emergency diesel generator and the high pressure core spray system were rendered inoperable. The operators also declared the division 3 SX system inoperable and unavailable. Before returning the breaker to the ON position, the licensee conducted troubleshooting and determined that the breaker had not tripped due to an actual adverse condition. The operators returned the breaker to its normal position after approximately one hour and declared the systems operable.

The licensee completed a root cause report and determined that the root cause of this event was that security management failed to adequately assess the inadvertent contact configuration control risk when the requirement for security officers to carry a protective mask in a bag attached to the thigh was implemented in 2005. The root cause also concluded that the licensee's Operations organization failed to provide adequate information and expectations to the plant organization regarding configuration control inadvertent contact events, at the level of detail needed to identify and mitigate these hazards.

Analysis: The inspectors determined that the failure to perform an adequate configuration control risk evaluation was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was more than minor in accordance with IMC 0612, "Power Reactor Inspection Reports," Appendix B, "Issue Screening," issued on November 17, 2006, because it impacted the equipment performance attribute of the mitigating systems cornerstone objective to ensure availability, reliability, and capability of systems that respond to initiating events. With the circuit breaker in the OFF position, 1SX014C would remain open during a loss of offsite power event. In this configuration, the SX system could not perform its safety function of supplying cooling water to both the division 3 diesel generator and the high pressure core spray (HPCS) pump room cooling system.

The inspectors completed a phase 1 significance determination using IMC 0609, "Significance Determination Process," Appendix A, Attachment 1, dated March 23, 2007. The inspectors concluded that the finding affected the mitigating systems cornerstone. The inspectors answered "Yes" to question 2, which asked whether this issue resulted in a loss of system safety function. The inspectors proceeded to do a phase 2 analysis and the results showed that this issue had very low safety significance (Green) due to the short duration exposure time, less than three days, and credit for operator actions to restore the system back to service. This finding involved the cross-cutting area of human performance (work practices). Licensee management failed to ensure the proper management and oversight of security personnel rounds activities.

Enforcement: No violation of regulatory requirements occurred. This issue was considered a finding of very low safety significance **(FIN 05000461/2007-002-02)**. This issue was entered into the licensee's corrective action program as IRs 520922 and 613643. Licensee Event Report 05000461/2006004-00, "Inadequate Configuration Control Risk Assessment Causes Loss of Safety Function" is closed.

.3 (Closed) LER 05000461/2006003-00: High Reactor Water Level Scram Result of Bad Inverter Circuit Board Solder Joint.

On August 27, 2006, Clinton Power Station experienced an automatic reactor scram on high water level condition (>52 inches). This high water level condition was created by the automatic start and injection of HPCS and a subsequent trip of the 1A reactor recirculation pump. The HPCS injection and recirculation pump trip occurred because of a failure of the Division 4 Nuclear Safety Protection System Inverter.

The licensee's investigation into this event determined that the inverter failure occurred because of an inadequate/faulty solder joint on a back plane connector. This conclusion was verified by the vendor and PowerLabs during post-event testing of the back plane connector. Both the vendor and PowerLabs concluded that poor quality soldering (lack of solder flow through etched circuit board eyelet) resulted in pooling (no penetration through eyelet) of solder at common connection between the R103 resistor, the J2 board, and the DC to DC converter.

In NRC Inspection Report 50-461/2006007, the inspectors documented a self-revealed Green finding and a Non-Cited Violation of 10 CFR Part 50, Appendix B, Criterion XVI, **(NCV 05000461/2006007-01)**. The NCV was identified due to the licensee's failure to identify and correct a condition adverse to quality. Specifically, the licensee failed to identify and correct the cause of a failure of the Division 4 Nuclear Safety Protection System (NSPS) Inverter in March of 2006. Licensee Event Report 05000461/2006003-00, "High Reactor Water Level Scram Result of Bad Inverter Circuit Board Solder Joint. " is closed.

4OA4 Cross-Cutting Aspects of Findings

.1 A finding described in Section 4OA3.2 of this report had, as its primary cause, a human performance deficiency, in that the licensee management failed to ensure proper management and oversight of security personnel rounds activities. This inadequate oversight resulted in the Division 3 SX system being rendered inoperable.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. B. Hanson and other members of licensee management at the conclusion of the inspection on April 5, 2007. The inspectors asked the licensee whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

.2 Interim Exit Meeting

An interim exit meeting was conducted for:

Maintenance Effectiveness Periodic Evaluation with Mr. B. Hanson, Site Vice President on January 26, 2007.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

B. Hanson, Site Vice President
R. Kearney, Plant Manager
R. Schenck, Work Management Director
G. Vickers, Radiation Protection Director
R. Frantz, Regulatory Assurance Representative
J. Icard, Maintenance Rule Coordinator
K. Scott, Regulatory Assurance Director
C. Vanderburgh, Nuclear Oversight Manager
J. Domitrovich, Maintenance Director
D. Schavey, Operations Director
W. Scott, Chemistry Manager
J. Lindsay, Training Director
C. Williamson, Security Manager
R. Peak, Site Engineering Director
T. Chalmers, Shift Operations Superintendent

LIST OF ITEMS OPENED, CLOSED AND DISCUSSED

Opened and Closed

05000461/2007-002-01	NCV	Failure of the electrical circuit card resulted in a loss of safety function for the main turbine bypass valves.
05000461/2007-002-02	FIN	Failure to perform an adequate configuration control risk evaluation was a performance deficiency warranting a significance evaluation.
05000461/2006002-00 and 01	LER	Main Turbine Bypass System Safety Function Lost Due to Circuit Card Failure
05000461/2006004-00	LER	Inadequate Configuration Control Risk Assessment Causes Loss of Safety Function.
05000461/2006003-00	LER	High Reactor Water Level Scram Result of Bad Inverter Circuit Board Solder Joint.

Discussed

None

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 pat of it, unless this is stated in the body of the inspection report.

<u>1R04</u> <u>Equipment Alignments</u>

CPS 3312.01E001, “Residual Heat Removal Electrical Lineup,” Revision 14
CPS 3312.01V001, “Residual Heat Removal Valve Lineup,” Revision 16a
M05-1075, P&ID Residual Heat Removal, sheets 2-4
AR 00607147 Report; 1VY01S: Temp Berm Allows Water To Leak Onto Floor.
AR 00607149 Report; 1VY02S: Temp Berm Allows Water To Leak Onto Floor.
AR 00607152 Report; 1VY03S: Temp Berm Allows Water To Leak Onto Floor.
AR 00607153 Report; 1VY05S: Temp Berm Allows Water To Leak Onto Floor.
AR 00607169 Report; Temperature Switch for RHR A Room Fan Is near Max Tolerance.
AR 00607176 Report; NRC Questions.
AR 00608028 Report; Scaffold Pole Contacting Insulation On SX Pipe.
AR 00607139 Report; 1E21C001: LPCS Pump Motor Lower Oil Level Needs Oil Added.
CPS 3313.01E001, Low pressure core spray electrical lineup; Rev 11a
CPS 3313.01V001, Low pressure core spray valve lineup; Rev 13
CPS 3313.01V002, Low pressure core spray instrument valve lineup; Rev 8a
CPS 3310.01E001, Reactor core isolation cooling electrical lineup; Rev 14b
CPS 3310.01V001, Reactor core isolation cooling valve lineup; Rev 12c
IR 519415, Handswitch 0HS-VC107 found in pull-to-lock; August 13, 2006
IR 523069 As found position of valve 1WO278B in question; August 23, 2006
IR 556498, Implementation of fleet configuration control actions; November 10, 2006

<u>1R05</u> <u>Fire Protection</u>

USAR Appendix E, Section 3.3.2.4 Fire area C-2, Elevation 778' 0" Containment, Revision 11
USAR Appendix E, Section 3.3.2.6 Fire area C-2, Elevation 803' 3" Containment, Revision 11
USAR Appendix E, Section 3.3.2.8 Fire area C-2, Elevation 828' 3" Containment, Revision 11
USAR Appendix E, Fire protection evaluation report, Rev 11
USAR Figure FP-2a, Fire zone boundaries auxiliary, fuel building and containment basement floor plan - EL 707'-6" & 712'-0", Rev 10
USAR Figure FP-2b, Fire protection features auxiliary, fuel building and containment basement floor plan - EL 707'-6" & 712'-0", Rev 10
AR 00596302 Report; Duct Tape Found On CO2 Nozzle In Div 1 DG Room
AR 00604868 Report: Improper Use Of RP Stanchions In Containment
AR 00605029 Report; NRC Weekly Meeting Follow-Up Item
AR 00607547 Report; NRC IDD - Walk Down Of Loose Grating Clips
USAR 3.8.1.3, Fire zone R-1c, elevation 702'-0" & 720'-6" general access area, Rev 11
USAR, Figure FP-16a, Fire zone boundaries radioactive waste building floor plan 702'-0", Rev 8

USAR, Figure FP-16b, Fire protection features radioactive waste building floor plan 702'-0", Rev 8
USAR, Figure FP-17a, Fire zone boundaries radioactive waste building intermediate floor plan elevation 720'-6"; Rev 7
USAR, Figure FP-17b, Fire protection features radioactive waste building intermediate floor plan elevation 20'-6"; Rev 7
USAR Appendix E, Sections 3.4.1.2, 3.4.1.3, 3.4.1.5, 3.4.1.6; Revision 11
USAR Appendix E Section 3.5.6, Fire Area D-6, Revision 11
USAR Appendix E Section 3.1.2.2.9, Fire Doors, Revision 11
USAR Appendix E Section 3.4.3.1, Fire zone CB-3a; Revision 11
CPS 1893.01, Fire Protection Impairment Reporting, Page 42 of 43; Revision 15e.

1R12 Maintenance Effectiveness (71111.12B)

ER-AA-310; Implementation of the Maintenance Rule; Revision 5
ER-AA-310-1005; Maintenance Rule - Dispositioning Between (a)(1) and (a)(2); Revision 4
ER-AA-310-1007; Maintenance Rule - Periodic (a)(3) Assessment; Revision 3
Assessment of Maintenance Effectiveness 10 CFR 50.65(a)(3) Assessment (3/1/2004 - 3/1/2006; dated May 31, 2006
Systems/Structures Scoping and Performance Criteria Clinton Power Station; dated December 15, 2006
Unavailability Status Cycle 10; dated December 5, 2006
Reliability Status; dated December 5, 2006
List of Cycle 10 Functional Failures: 3/1/2004 - 3/1/2006; dated December 5, 2006
Maintenance Rule Expert Panel Meeting Minutes; dated May 2003 through December 2006
FASA No. 462465; Clinton Station-Maintenance Rule Program; dated August 1, 2006
System Health Overview Report; VC - Control Room Ventilation; 2nd Quarter 2006
(a)(1) Action Plan - Development and Monitoring Goal Setting Template - Leak Detection; dated May 3, 2004
(a)(1) Action Plan - Development and Monitoring Goal Setting Template - Control Room Ventilation; Revision 3; dated January 16, 2007
(a)(1) Action Plan Return to (a)(2) - Neutron Monitoring; dated May 20, 2004
AR00265602; Indicated Rise in Unidentified Drywell Leakage; dated October 21, 2004
AR00375941; Reactive Load Swings During diesel generator Operability; dated September 21, 2005
AR00387034; LPRM 1C51K605DD Read Incorrectly due to Aging; dated October 19, 2005
AR00437890; 0VC13CA Received Unexpected MCR Annunciator 5050-26 (CR Ventilation); dated December 31, 2005

1R15 Operability Evaluations

IR 589461, Found reactor core isolation cooling turbine and governor oil below the standby band; February 9, 2007
IR 563348, Minor oil leaks on oil sightglass and governor assembly; November 29, 2006
IR 582162, 1DG01FC: Loose bolt on division 2 emergency diesel generator air intake filter housing; January 23, 2007
IR 582173, 1DG01FE: Loose bolt on division 1 emergency diesel generator air intake filter housing; January 23, 2007

CR 2-01-01-206, As found mounting configuration for division 2 diesel generator air filter differs from seismic qualified configuration; January 24, 2001
ECN 32414, Accept the as found mounting configuration on the division 2 diesel generator combustion air filter intake unit; January 18, 2001
ECR 379228, Reactor core isolation cooling turbine oil level 1/8" below standby level I the sight glass, IR 589461; February 9, 2007
EC 334968, Engineering evaluation assignment number 4 for CR-91417; January 28, 2002
WO 974880, Visually inspect trip throttle valve linkage and sample turbine oil; February 12, 2007
WO 980558, Minor oil leaks on oil sightglass and governor assembly; November 30, 2006

1R22 Surveillance Testing

CPS 9051.02, High pressure core spray valve operability test; Revision 38d
C NSED IP-C-0084 Revision 0; Setpoint Calculation for ATWS RPV Low-Low Level Trip; 1B21N400A,B,E,F.
Calculation No. IP-C-0084 Attachment 1; Scaling of the ATWS RPV Low-Low Level 2 Trip Loop.
9434.01, Revision 034 F, ATWS Reactor Vessel Water Level B21-N400A(B,E,F) Channel Calibration.
9434.01D001, Revision 034, A, ATWS Reactor Vessel Water Level B21-N400A Channel Calibration Data SHT 02PS082.
AR 00606797 Report; Abnormal Noise In Containment During Stroking of 1E12F028B.
AR 00607139 Report; 1E21C001 LPCS Pump Motor Lower Oil Level Needs Oil Added.
AR 00594949 Report; ATWS Level Channels A/B/F Are Reading › Other Instruments.
AR 00597859 Report; ATWS LVL Channel Check Per TS3.3.4.2.1 NRC Identified.
AR 00607833 Report; RHR B System Pressure Higher In Standby.
AR 00595476 Report; During performance on 9434.01 Inop Light Would Not Clear.
CPS 9052.01, Low pressure core spray/residual heat removal A pumps and low pressure core spray/residual heat removal A water leg pump operability; Rev 43d
WO 974878, 9052.01R21 Low pressure core spray pump operability; January 30, 2007
WO 973336, 9052.01S21 Low pressure core spray/residual heat removal A water leg pump operability; January 30, 2007
IR 552738, Typographical error in CPS 9052.01D001; November 2, 2006
9080.03D001, Revision 021 B, Diesel Generator 1C Operability - Manual and Quickstart Data Sheet 05PS402.
9080.01, Revision 050 D, Diesel Generator 1A Operability - Manual and Quickstart Operability.
9080.01D001, Revision 043 E, Diesel Generator 1A Operability - Manual and Quickstart Data Sheet.
9080.02, Revision 047 B, Diesel Generator 1B Operability - Manual and Quickstart Operability.
9080.02D001, Revision 041 D, Diesel Generator 1B Operability - Manual and Quickstart Data Sheet 05PS 124.
9080.03, Revision 028 C, Diesel Generator 1C Operability - Manual and Quickstart Operability.
AR 00582162 Report; 1DG01FC: Loose Bolt on Div 2 EDG Air Intake Filter Housing.
AR 00582671 Report; 1DG619PA - Div 3 DG Main Lube Oil Pump Oil Leak When Started.
AR 00585986 Report; 1PIDG306A: Out of Spec Readings on Div 1 DG Data.
AR 00594351 Report; Procedure Guidance For DG Fuel system Alarm Lacking.
AR 00594364 Report; 1DG01KC NRC Identified Questions From Monthly Div 3 EDG Run.
AR 00594398 Report; Div 3 DG Starting Air Pipe Hanger, Not Supporting Pipe.

AR 00596974 Report; Replace Div 1 DG Air Dryer Moisture Indicator Dessicant.
AR 00596982 Report; Replace Div 2 DG Air Dryer Moisture Indicator Dessicant.
AR 00500452 Report; Jacket Water level on Both 12 and 16 Cylinder Engines High.

1R23 Temporary Modifications

EC 365082, Revision 0; Disconnect leads for circuit breaker 252-AT1AA1 relay 74-AT1AA1 to bypass alarm to window 1D on main control room panel 1H13-P877-14A.

4OA2 Identification and Resolution of Problems

IR 585504, Main control room differential pressure not acceptable after shifting to high-rad mode; January 31, 2007
IR 324015, Scheduling scaffold preparations for division 2 diesel generator outage; April 12, 2005
IR 365967, Fuel handling bridge came in contact with a scaffold pole; August 23, 2005
IR 396679, Hanger supporting instrument air line from a fire protection line; November 9, 2005
IR 448776, 1B21N808 broken flex conduit safety relief valve acoustic monitor in drywell; February 1, 2006
IR 455915, Scaffold interferes with permanent plant equipment; February 20, 2006
IR 456335, Damaged piping insulation; February 20, 2006
IR 309311, Unexpected alarm 5000-3F during chill water pump shift; March 7, 2005
IR 302834, Step change in reactor water cleanup heat exchanger room west differential temperature; February 18, 2005
IR 256599, Received main control room annunciator 5000-3F reactor water cleanup equipment room differential temperature high; September 24, 2004
IR 260604, Main control room alarm 5000-3F ‘A’ reactor water cleanup heat exchanger area high differential temperature, safety parameter display system EOP-8 alarm; October 6, 2004
IR 256368, Containment equipment drain sump inleakage increasing trend; September 9, 2004
IR 148929, Reactor water cleanup equipment room high differential temperature; March 13, 2003
IR 573291, Shift of chilled water pumps causes safety parameter display system alarm / EOP evaluation; December 27, 2006
IR 581980, Reactor water cleanup A heat exchanger room delta temperature high alarm 5000-3F, 5004-3F; January 23, 2007
IR 583992, Reactor water cleanup A heat exchanger room delta temperature high alarm 5000-3F, 5004-3F; January 26, 2007
IR 585497, Received main control room annunciator 5004-3F safety parameter display system critical safety feature alarm; January 31, 2007
IR 587047, Thermography identified seat leakage past 1G33F018A/19A; February 2, 2007
IR 587777, Delays encountered in shifting reactor water cleanup heat exchanger trains; February 5, 2007

4OA3 Event Followup

AR 00613643 Report: No Procedural Process For Developing PCM Templates.

LIST OF ACRONYMS USED

ADAMS	Agency wide Documents Access and Management System
CSF	Critical Safety function
DCA	Direct Current Alarm
DG	Diesel Generator
EACE	Equipment Apparent Cause Evaluation
IMC	Inspection Manual Chapter
IR	Issue Reports
LCO	Limiting Condition of Operation
LD	Leak Detection
MR	Maintenance Rule
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NR	Neutron Monitoring
NRC	Nuclear Regulatory Commission
PARS	Publicly Available Records
PCM	Performance Centered Maintenance
PM	Preventative Maintenance
PMCE1	Plant Material Condition Excellence Initiative
RI	Reactor Core Isolation Cooling
SDP	Significant Determination Process
SPDS	Safety Parameter Display System
SSC	Structures, Systems, and Components
SX	Service Water
TRM	Technical Requirements Manual
TS	Technical Specification
USAR	Updated Safety Analysis Report
VC	Control Room Ventilation